Final Submittal

(Blue Paper)

CATAWBA 2008-301 -
SRO Written EXAMINATION

12 - 10 - 2008

FINAL SRO

WRITTEN EXAMINATION

AND REFERENCES

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date: 12-10-2008	Facility/Unit: CATAWBA 2008-301
Region: I (II) III IV	Reactor Type: (W) CE BW GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO port

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

_____
Applicant's Signature

Results		
RO/SRO-Only/Total Examination Values	____ / ____ / ____	Points
Applicant's Score	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

SRO MASTER COPY

MARKED UP

CATAWBA NUCLEAR STATION

2008 SRO NRC Examination

Question: 1
(1 point)

Unit 1 is at 100% reactor power.

Four hours ago:

- PZR Level Select Switch was in the 3-2 position
- PZR level channel 1 failed HIGH
- All actions required by Technical Specifications were completed to allow continued unit operation.

Following the receipt of several annunciators, the following items are noted:

- 1EDC has lost power
- 1FO-1, B/6 (PZR Hi Level RX Trip) is LIT and RED
- DRPI indicates control bank position at 215 steps on Bank D
- Both RX TRIP BKR 1A and 1B red lights are LIT.

Which one of the following describes:

1. The current condition of the plant and
2. The correct operator action to take for the above evolution?

A. 1. Anticipated Transient Without Scram (ATWS)
   2. Manually trip the reactor

B. 1. Anticipated Transient Without Scram (ATWS)
   2. Perform a shutdown per OP/1/A/6100/003 (Controlling Procedure for Unit Operation)

C. 1. Reactor Protection System (RPS) failure
   2. Manually trip the reactor

D. 1. Reactor Protection System (RPS) failure
   2. Perform a shutdown per OP/1/A/6100/003 (Controlling Procedure for Unit Operation)

Question: 2
(1 point)

Initial Conditions

- Unit 1 was in Mode 3 cooling down for a refueling outage per OP/1/A/6100/002 (Controlling Procedure for Unit Shutdown)
- NC pressure is 1500 psig
- NC temperature is 500°F and slowly decreasing

Operators note the following:

- 1RAD-1, B/3 "1EMF41 AUX BLDG VENT HI RAD" - LIT
- 1AD-13, A/1 "ND & NS ROOMS SUMP LEVEL EMERG HI" - LIT
- "SAFETY INJECTION ACTUATED" status light - LIT

Which one of the following states the correct procedure flowpath that will address this event?

A. AP/1/A/5500/027 (Shutdown LOCA)
AP/1/A/5500/019 (Loss of Residual Heat Removal System)

B. AP/1/A/5500/027 (Shutdown LOCA)
AP/1/A/5500/010 (Reactor Coolant Leak)

C. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)
EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant)
EP/1/A/5000/ES-1.2 (Post LOCA Cooldown and Depressurization)

D. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)
EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant)
EP/1/A/5000/ECA-1.2 (LOCA Outside Containment)

Question: 3
(1 point)

Given the following events:

- A Large Break LOCA has occurred on Unit 2
- All equipment functioned as designed
- The OSM has declared an Alert
- A signed Emergency Notification Sheet has been handed to you for transmittal

Which of the following is a complete list of agencies required to be contacted within 15 minutes of the declaration of the Alert?

A. State and county warning points and the NRC Operations Center

B. County warning points and NRC Operations Center

C. State warning points and NRC Operations Center

D. State and county warning points

Question: 4
(1 point)

Given the following initial conditions:

- 1NV-294 (NV Pmps A&B Disch Flow Ctrl) in MANUAL
- 1NV-309 (Seal Water Injection Flow) in MANUAL
- pressurizer pressure is 2235 psig
- total seal water flow is 32 gpm
- charging line flow is 89 gpm

If pressurizer pressure is increased to 2300 psig, which one of the following sets of system parameter changes is correct?

A. Charging line flow decreases and total seal water flow decreases

B. Charging line flow decreases and total seal water flow remains the same

C. Charging pump discharge header pressure increases and total seal water flow increases

D. Charging pump discharge header pressure increases and total seal water flow remains the same

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Question: 5
(1 point)

Unit 1 was in Mode 5 preparing to enter Mode 6.

Given the following:

- Both trains of ND have been lost.
- The crew entered AP/1/A/5500/019 (Loss of Residual Heat Removal System) but actions to restore cooling have failed.
- The OSM has determined an immediate need to take an action per 10CFR50.54(X).

Per the requirements of OMP 1-7 (Emergency/Abnormal Procedure Implementation Guidelines):

1. Is notification to the NRC Operations Center required prior to taking the action?
2. How many additional SROs (if any) are required to agree with the OSM prior to the action being taken?

A. 1. Yes
   2. None

B. 1. Yes
   2. One additional SRO

C. 1. No
   2. None

D. 1. No
   2. One additional SRO

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Question: 6
(1 point)

Unit 1 was operating at 100% with "A" Train KC in service. Given the following:

- An 86N relay actuated on 1ETB two minutes ago
- A major KC system piping leak has occurred in the Auxiliary Building non-essential header
- 1AD-10, A/1 "KC SURGE TANK A LO-LO LEVEL" - LIT
- 1AD-10, A/2 "KC SURGE TANK B LO-LO LEVEL" - LIT
- The crew has entered AP/1/A/5500/021 (Loss Of Component Cooling)

Assuming all automatic actions have occurred, which one of the following correctly lists the major KC headers that are currently being cooled?

A. KC Train A essential header only

B. KC Train A essential header and the Reactor Building non-essential header

C. KC Train A essential header and KC Train B essential header

D. KC Train A essential header, KC Train B essential header and the Reactor Building non-essential header

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Question: 7
(1 point)

Given the following:

- The SSF has been manned due to a fire in the cable spreading room.
- During the course of SSF operations a head vent stuck in the open position for a short period of time and then reclosed.
- You have been directed to increase NC pressure using heaters.

1. Why is pressure recovery slower from the SSF than from the Control Room?
2. How are the heaters available from the SSF secured should Pzr level drop below 17%?

A. 1. Only a portion of the D heaters are available from the SSF
   2. Automatically

B. 1. Only a portion of the D heaters are available from the SSF
   2. Manually

C. 1. Only A and B heaters are available from the SSF
   2. Automatically

D. 1. Only A and B heaters are available from the SSF
   2. Manually

Question: 8
(1 point)

Which one of the following is a complete list of breakers directed to be opened per EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS) to trip the reactor locally?

1. Reactor Trip Breakers RTA and RTB
2. Reactor Trip Bypass Breakers BYA and BYB
3. CRD/MG "Motor" Breakers
4. CRD/MG "Generator" Breakers

A. 1 and 2 only

B. 1, 2, and 3 only

C. 1, 2, and 4 only

D. 1, 2, 3, and 4

Question: 9
(1 point)

Given the following:

- Unit 1 and 2 are operating at 100%
- One single steam generator tube fully shears on each unit
- The crews are responding per EP/1(2)/A/5000/E-3 (Steam Generator Tube Rupture), preparing to perform the initial reactor coolant system cooldown to the required core exit thermocouple temperature using steam dumps.

Based on the differences between Unit 1 and Unit 2 steam generator design:

1. Which unit would have a lower primary system equilibrium pressure?
2. Which unit will have a faster cooldown rate?

(Assume identical cores and steam dump performance.)

A. Unit 1 would have a lower equilibrium pressure and Unit 1 would have a faster cooldown rate.

B. Unit 1 would have a lower equilibrium pressure and Unit 2 would have a faster cooldown rate.

C. Unit 2 would have a lower equilibrium pressure and Unit 1 would have a faster cooldown rate.

D. Unit 2 would have a lower equilibrium pressure and Unit 2 would have a faster cooldown rate.

Question: 10
(1 point)

Unit 1 is operating at 100%. 1ERPA is lost. What effect does this have on VCT auto makeup capability and VCT level indication in the control room?

	Auto Makeup	Level Indication
A.	Available	Available
B.	Unavailable	Available
C.	Available	Unavailable
D.	Unavailable	Unavailable

Question: 11
(1 point)

Given the following:

- 1ERPD has been de-energized due to a blown fuse on inverter 1EID.
- The crew has implemented AP/1/A/5500/029 (Loss of Vital or Aux Control Power).
- The fuse has been replaced and the CRS wishes to re-energize 1ERPD from 1EID.

Per OP/1/A/6350/008 (125VDC/120VAC Vital Instrument and Control Power System), which one of the following correctly states the minimum acceptable wait time prior to inverter restart and the sequence for operation of inverter 1EID DC input breaker and AC output breaker?

A. 1. 5 seconds
   2. Close the DC input breaker and then close the AC output breaker

B. 1. 5 seconds
   2. Close the AC output breaker and then close the DC input breaker

C. 1. 60 seconds
   2. Close the DC input breaker and then close the AC output breaker

D. 1. 60 seconds
   2. Close the AC output breaker and then close the DC input breaker

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Question: 12
(1 point)

Both units were at 100% with 2A RN Pump in service when the following annunciators were received:

- 1AD-12, E/2 "RN PIT A SWAP TO SNSWP" - LIT
- 2AD-12, E/2 "RN PIT A SWAP TO SNSWP" - LIT
- 1AD-12, B/1 "RN PUMP INTAKE PIT A LEVEL - LO" - LIT
- 2AD-12, B/1 "RN PUMP INTAKE PIT A LEVEL - LO" - LIT

What is the minimum time the crew must wait following receipt of these annunciators prior to operating RN equipment per AP/0/A/5500/020 (Loss of Nuclear Service Water) and what is the reason for that time delay?

A. 2 minutes;
To allow sufficient time for all components to respond and allows the operator an opportunity to verify the signal is valid prior to any system realignments.

B. 2 minutes;
To prevent an automatic swap to the pond if RN pit level can be restored within 2 minutes.

C. 5 minutes;
To allow sufficient time for all components to respond and allows the operator an opportunity to verify the signal is valid prior to any system realignments.

D. 5 minutes;
To prevent an automatic swap to the pond if RN pit level can be restored within 5 minutes.

Question: 13
(1 point)

Given the following:

- One RL turnaround valve is manually pinned in place for mantenance
- The crew has entered AP/0/A/5500/022 (Loss of Instrument Air)
- Operators have determined that the leak can be isolated but doing so will result in all RL turnaround valves losing VI.
- The CRS has directed that the leak be isolated.

Which one of the following correctly states the effect that this will have on the RL turnaround valves and the equipment cooled by RL.

A. The unpinned RL turnaround valves will fail open resulting in more flow to the components supplied by RL.

B. The unpinned RL turnaround valves will fail closed resulting in more flow to the components supplied by RL.

C. The unpinned RL turnaround valves will fail open resulting in less flow to the components supplied by RL.

D. The unpinned RL turnaround valves will fail closed resulting in less flow to the components supplied by RL.

Question: 14
(1 point)

Given the following:

- Unit 1 is at 100% power with power factor at 0.99 lagging.
- Operators are controlling power factor in manual due to the auto voltage regulator not controlling properly.
- A major grid disturbance causes power factor to increase to slightly leading.

1. Which button on the voltage regulator is operated to bring power factor back to its original value?
2. What part of the generator is susceptible to overheating should power factor be erroneously adjusted to 0.8 lagging?

Reference provided

A. 1. The "LOWER" button
   2. The generator armature core end

B. 1. The "RAISE" button
   2. The generator armature core end

C. 1. The "LOWER" button
   2. The generator field

D. 1. The "RAISE" button
   2. The generator field

Question: 15
(1 point)

Unit 1 was operating at 100%. Given the following events and conditions:

- 0200 – reactor tripped due to a LOCA outside containment
- 0210 – crew enters ECA-1.2, (LOCA Outside Containment)
- 0220 – crew enters ECA-1.1, (Loss of Emergency Coolant Recirc)
- 0240 – The crew is at the step in ECA-1.1 to determine NC subcooling
- Current conditions:
  - NCS pressure is 1100 psig
  - 1B NC pump running
  - 1A, 1C, and 1D NC pumps secured
  - Reactor Vessel D/P is 20%
  - 1 NI pump running, indicating 220 gpm
  - 1 NV pump running, indicating 385 gpm
  - Both ND pumps off
  - No NS pumps running
  - Subcooling is 35°F

Which one of the following statements correctly describes the <u>minimum</u> required flow and which pump can be secured?

<u>Reference provided</u>

A. 210 gpm, stop the running NV pump.

B. 210 gpm, stop the running NI pump.

C. 410 gpm, stop the running NI pump.

D. 410 gpm, neither pump may be secured at this time.

Question: 16
(1 point)

A feedwater transient resulted in a reactor trip and the operating crew entered EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink) when all Auxiliary Feedwater flow was lost. Given the following:

- S/G 1A wide range level – 31%
- S/G 1B wide range level – 20%
- S/G 1C wide range level – 23%
- S/G 1D wide range level – 28%
- The BOP has <u>just</u> secured all the NC pumps
- The OATC notes NC system pressure is increasing

1. Why have NC pumps been secured?
2. Why is NCS pressure increasing?

A.
1. To begin NCS bleed and feed
2. Due to NC temperature increase

B.
1. To minimize heat input
2. Due to letdown being secured

C.
1. To begin NCS bleed and feed
2. Due to letdown being secured

D.
1. To minimize heat input
2. Due to NC temperature increase

Question: 17
(1 point)

The crew implemented EP/1/A/5000/ECA-1.2 (LOCA Outside Containment), determined the leak can not be isolated and transitioned to EP/1/A/5000/ECA-1.1 (Loss of Emergency Coolant Recirculation). Given the following:

- FWST level is 55%
- Subcooling is +7°F.

What actions, if any, are taken per EP/1/A/5000/ECA-1.1 to ensure the NV pumps maintain adequate suction until cold leg recirculation capability is restored?

A. Terminate safety injection and establish normal charging from the VCT.

B. Remove power from 1NI-184B (ND Pump 1B Cont Sump Suct) and 1NI-185A (ND Pump 1A Cont Sump Suct)

C. Use "DEFEAT" buttons for "C-LEG RECIR FWST TO CONT SUMP SWAP TRN A" and "C-LEG RECIR FWST TO CONT SUMP SWAP TRN B"

D. None, a swap to the containment sump is blocked when sump level is less than 3.3 feet

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Question: 18
(1 point)

The crew entered EP/1/A/5000/ECA-2.1 (Uncontrolled Depressurization of All Steam Generators) following a unit trip. Given the following:

- Attempts to close any MSIV using its individual valve control board pushbutton have failed.
- Safety Injection has not been reset.
- 1AD-03, C/5 "SM ISOL TRN A" - LIT
- 1AD-03, D/5 "SM ISOL TRN B" – LIT
- 1AD-03, E/5 "SM ISOL VLVS NOT FULLY OPEN" - DARK

1. What additional action is taken per this procedure to attempt to close any MSIV?
2. If an MSIV can be closed, what plant parameter is monitored to determine when this procedure can be exited?

A. 1. Maintenance is dispatched to isolate air to the MSIVs
   2. NC loop T-hots

B. 1. Both trains of Main Steam Isolation are manually initiated
   2. NC loop T-hots

C. 1. Maintenance is dispatched to isolate air to the MSIVs
   2. S/G pressure

D. 1. Both trains of Main Steam Isolation are manually initiated
   2. S/G pressure

Question: 19
(1 point)

Given the following events and conditions on Unit 1:

- NC system is at full temperature and pressure.
- "A" Shutdown Bank control rods are fully withdrawn.
- CRD BANK SELECT switch is in the "SBB" position.
- The OATC is withdrawing "B" Shutdown Bank control rods with the current bank position at 64 steps withdrawn.
- The OATC releases the ROD MOTION switch but "B" Shutdown Bank control rods continue to withdraw.

1. What is the current plant Mode of Operation?
2. Which of the following describes the first <u>required</u> action(s) for this situation per AP/1/A/5500/015 (Rod Control Malfunction)?

A. 1. Mode 2
   2. Immediately trip the reactor.

B. 1. Mode 3
   2. Immediately trip the reactor.

C. 1. Mode 2
   2. Immediately place CRD BANK SELECT switch IN MANUAL; if rods continue to move then trip the reactor.

D. 1 Mode 3
   2. Immediately place CRD BANK SELECT switch IN MANUAL; if rods continue to move then trip the reactor.

Question: 20
(1 point)

Unit 1 was operating at 100% power with Control Rod Bank D at 216 steps withdrawn on DRPI when an OTDT runback occurred for approximately 30 seconds and cleared.

When conditions stabilized, the following indications were noted:

- Control Rod Bank D demand counters are indicating 190 steps.
- Control Rod Bank D rod D4 indicates 216 steps withdrawn on DRPI.
- All other Control Rod Bank D rods indicate 188 steps withdrawn on DRPI.

1. What is the <u>first</u> immediate action of the Abnormal Procedure that will address this issue?
2. What are the modes of applicability for the corresponding Technical Specification?

A. 1. Verify only one rod – MISALIGNED.
   2. MODE 1, MODE 2 with $k_{eff} \geq 1.0$

B. 1. Verify only one rod – MISALIGNED.
   2. MODE 1, MODE 2

C. 1. Ensure "CRD BANK SELECT" switch – IN MANUAL.
   2. MODE 1, MODE 2 with $k_{eff} \geq 1.0$

D. 1. Ensure "CRD BANK SELECT" switch – IN MANUAL.
   2. MODE 1, MODE 2

Question: 21
(1 point)

Given the following:

- Unit 1 is in Mode 5
- BAT temperature is 60° F.
- FWST temperature is 70° F.

Assuming any required pumps are operable, which one of the following correctly states a combination of equipment which will satisfy the requirements of SLC 16.9-7 Boration System Flowpaths – Shutdown?

A. BAT to NV Pump

B. FWST to NI Pump via 2 cold leg lines

C. FWST to NV Pump

D. FWST to ND Pump via 2 cold leg lines

Question: 22
(1 point)

Given the following conditions and sequence of events:

- During the last calibration of N-35, an IAE technician improperly adjusted the compensating voltage to a value slightly lower than required by procedure.
- N-36 failed 3 hours ago, the crew entered AP/1/A/5500/016 (Malfunction of Nuclear Instrumentation), Case III (Intermediate Range Malfunction).
- All actions required by AP/1/A/5500/016 have been completed.
- A feedwater transient occurs resulting in a reactor trip.

How does this adjustment error affect the reading on N-35 and how will this condition affect when the source range instruments automatically energize?

A. N-35 will indicate higher than the actual value.
The source ranges instruments will energize at a lower actual neutron flux.

B. N-35 will indicate higher than the actual value.
The source ranges instruments will energize at the same actual neutron flux.

C. N-35 will indicate lower than the actual value.
The source ranges instruments will energize at the same actual neutron flux.

D. N-35 will indicate lower than the actual value.
The source ranges instruments will energize at a higher actual neutron flux.

Question: 23
(1 point)

Given the following:

- Unit 1 is operating with a known 0.6 GPD S/G tube leak
- 1A CF pump tripped and results in a plant runback.
- The crew has stabilized the plant at the runback target per AP/1/A/5500/003 (Load Rejection)
- The transient has caused the tube leak to increase to 12 GPD.

Which one of the following indications will provide the best indication (most sensitive and timely) that the S/G tube leak has increased?

A. Observing 1EMF-26, 27, 28 and 29 (Steamline 1A – 1D)

B. Comparing S/G feed flow to steam flow mismatch

C. Observing 1EMF-33 (Condenser Air Ejector Exhaust)

D. Observing 1EMF-71, 72, 73, 74 (S/G A-D leakage)

Question: 24
(1 point)

S/G depressurization to atmospheric pressure has been performed in EP/1/A/5000/FR-C.1 (Response to Inadequate Core Cooling).

1. What are the NC temperature and RVLIS level limits that allow the crew to transition out of this procedure?
2. Why are these conditions more restrictive than earlier transition conditions?

A. 1. Two NC Thots less than 328 deg F, RVLIS level greater than 41%
   2. To ensure a hard bubble does <u>not</u> block natural circulation flow

B. 1. Two NC Thots less than 328 deg F, RVLIS level greater than 41%
   2. Due to the NC system being depressurized

C. 1. Two NC Thots less than 350 deg F, RVLIS level greater than 61%
   2. To ensure a hard bubble does <u>not</u> block natural circulation flow

D. 1. Two NC Thots less than 350 deg F, RVLIS level greater than 61%
   2. Due to the NC system being depressurized

Question: 25
(1 point)

Unit 1 was conducting refueling operations in mode 6. Given the following events and conditions:

- The containment purge (VP) system is in operation in the REFUEL mode.
- Both trains of SSPS are in "TEST".
- The refueling crew dropped a fuel assembly into the refueling cavity.
- 1RAD-1 A/2 "1EMF-39 CONTAINMENT GAS HI RAD" - LIT
- 1RAD-3 D/2 "1EMF-17 REACTOR BLDG REFUEL BRIDGE" - LIT
- The crew has implemented AP/1/A/5500/025 (Damaged Spent Fuel).

1. Based on the above conditions, what was the status of the VP system when AP/1/A/5500/025 was entered?
2. What is the reason for establishing closure prior to VP being secured?

A. 1. The VP system was running
   2. To prevent an unmonitored release

B. 1. The VP system was running
   2. To prevent an excessive negative pressure in containment

C. 1. The VP system has tripped
   2. To prevent an unmonitored release

D. 1. The VP system has tripped
   2. To prevent an excessive negative pressure in containment

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Question: 26
(1 point)

Unit 1 was operating at 100% power when a small break LOCA occurred. Given the following events and conditions:

- Cooldown and depressurization is in progress in ES-1.2 (Post Cooldown and Depressurization)
- NC system pressure has stabilized at 410 psig
- NC temperature has stabilized at 325°F
- FWST level is 70% and slowly decreasing
- The operators attempt to place 1A ND train in the RHR mode
- 1ND-1B and 1ND-2A (ND Pump 1A Suct from Loop B) will <u>not</u> open

Which one of the following statements correctly describes why 1ND-1B and 1ND-2A will <u>not</u> open?

A. ECCS has <u>not</u> been reset

B. The NC system pressure is too high

C. 1NI-147B (NI Pumps Recirc to FWST Isol) is open

D. 1NI-185A (ND pump 1A Suct from CNMT Sump) is closed

Question: 27
(1 point)

Given the following conditions and sequence of events:

- One hour ago, a fault in the Unit 1 main generator resulted in a complete loss of offsite power.
- The crew entered EP/1/A/5000/ES-0.2 (Natural Circulation Cooldown).
- The OSM determined that a transition to EP/1/A/5000/ES-0.3 (Natural Circulation Cooldown With Steam Void in Vessel) was required.
- The crew has transitioned to ES-0.3 and is preparing to depressurize the NC system.

1. What condition would require stopping the depressurization of the NC system during this cooldown?
2. What is the basis for stopping the depressurization?

A. 1. PZR Level greater than 70%
   2. To prevent loss of natural circulation

B. 1. RVLIS Level less than 73%
   2. To prevent loss of natural circulation

C. 1. PZR Level greater than 70%
   2. To ensure normal pressurizer pressure control response

D. 1. RVLIS Level less than 73%
   2. To ensure normal pressurizer pressure control response

Question: 28
(1 point)

Unit 1 is in the process of performing a reactor startup. Given the following conditions and sequence of events:

- Control Bank "A" is at 28 steps withdrawn
- 1AD-6, A/5 "NCP HI VIBRATION" - LIT
- 1AD-6, B/5 "NCP HI-HI VIBRATION" - LIT
- The BOP validates that the 1C NC Pump vibration level on the frame is at 6.5 mils using the NC Pump vibration monitor panel.

Which one of the following selections is the list of the correct actions based on this situation?

A. Trip 1C NC Pump.
Go to AP/1/A/5500/004 (Loss of Reactor Coolant Pump).

B. Reinsert Control Bank "A" rods.
Trip 1C NC Pump.
Go to AP/1/A/5500/004 (Loss of Reactor Coolant Pump).

C. Pump trip criteria is not yet met.
Go To AP/1/A/5500/008 (Reactor Coolant Pump Malfunction).

D. Trip the reactor.
Trip 1C NC Pump.
Go to EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Question: 29
(1 point)

Unit 1 is at 75% power and decreasing in preparation for entering a refueling outage. Given the following conditions and sequence of events:

- There is confirmed failed fuel on Unit 1.
- 1AD-07, F/3 "LETDN HX OUTLET HI TEMP" - LIT
- The BOP notes that letdown temperature has trended to 132°F and appears to have stabilized.

1. What minimum actions are required to reduce activity level per AP/1/A/5500/018 (High Activity in Reactor Coolant)?
2. What is the applicability of Tech Spec 3.4.16 (RCS Specific Activity)?

A. 1. Ensure at least one mixed bed demineralizer in service only.
   2. Modes 1, 2, and 3.

B. 1. Ensure at least one mixed bed demineralizer in service only.
   2. Modes 1 and 2, Mode 3 with Tavg $\geq$ 500°F.

C. 1. Reduce letdown temperature to clear the alarm and then place additional demineralizers in service.
   2. Modes 1, 2, and 3.

D. 1. Reduce letdown temperature to clear the alarm and then place additional demineralizers in service.
   2. Modes 1 and 2, Mode 3 with Tavg $\geq$ 500°F.

Question: 30
(1 point)

Unit 1 is operating at 100%. Given the following initial conditions and sequence of events:

- Excess letdown is in service to the VCT to repair a leak on the letdown line.
- A PZR pressure channel failure causes 1NC-32B (PZR PORV) and 1NC-36B (PZR PORV) to open.
- 1NC-36B does not re-close and the BOP closed its isolation valve.
- Minimum NC pressure reached during the event was 1820 psig.
- Current NC pressure is 2145 psig and increasing.

Assuming no operator actions other than isolating 1NC-36B:

1. What tank other than the VCT can excess letdown be directed to by 1NV-125B (Excess Letdn Hx Otlt Ctrl)?
2. Is excess letdown currently flowing to the VCT?

A. PRT; no

B. PRT; yes

C. NCDT; no

D. NCDT; yes

Question: 31
(1 point)

1ND-1B (ND Pump 1A Suct Frm Loop B) and 1ND-37A (ND Pump 1B Suct Frm Loop C) have been aligned to their alternate power supplies.

1. What impact (if any) will aligning the alternate power supply have on the interlocks associated with these valves?
2. How are these valves positioned electrically in the current alignment?

A. 1. Interlocks operate normally
   2. From the main control boards

B. 1. Interlocks operate normally
   2. From the face of the alternate MCC breaker

C. 1. Interlocks are removed
   2. From the main control boards

D. 1. Interlocks are removed
   2. From the face of the alternate MCC breaker

Question: 32
(1 point)

At 1200, Unit 1 was addressing an NC system leak per AP/1/A/5500/010 (Reactor Coolant Leak) when the leak began to increase. Given the following:

Time	1200	1206	1212	1218	1224
NC system pressure (psig)	2130	1950	5	5	5
Containment pressure (psig)	0.5	1.3	2.8	4.2	2.5
FWST level (%)	98	97	80	60	35

What is the earliest time that KC flow is automatically aligned to the ND heat exchangers?

A. 1206

B. 1212

C. 1218

D. 1224

Question: 33
(1 point)

The crew is performing actions of AP/1/A/5500/010 (Reactor Coolant Leak) due to an increase in charging flow required to maintain pressurizer level.

You have just completed an evaluation of PRT conditions and noted the following:

- PRT pressure is 12 psig and slowly increasing
- PRT temperature is 140°F and slowly increasing

The CRS directs you to monitor inputs to the PRT per Enclosure 13 (Possible NC System Leakage Paths to PRT).

Assuming a <u>single</u> valve is leaking by its seat, which valve could have caused the noted PRT indications?

A. 1NC-5 (Loop A Lo Point Drn)

B. 1NC-250A (Rx Head Vent Block)

C. 1NC-25A (Rx Head Gasket Leakoff Isol)

D. 1NV-87 (NC Pumps Seal Return Hdr Inside Relief)

Question: 34
(1 point)

Unit 1 is in Mode 3 with all shutdown banks withdrawn in preparation for startup when the following occur:

- 1AD-6 E/3 "NCP THERMAL BARRIER KC OUTLET HI/LO FLOW" - LIT
- OAC indicates KC flow to NCP 1C Thermal Barrier HX is 75 gpm.

What effects will this have on NCP 1C and what action should be taken to address the alarm?

A. NCP 1C seal cooling is being maintained. Verify 1KC-345A (NC Pump 1C Therm Bar Otlt) closes after a 30 second time delay.

B. NCP 1C seal cooling is being maintained. Verify 1KC-345A (NC Pump 1C Therm Bar Otlt) closes immediately.

C. All seal cooling to NCP 1C is lost. Open the #1 seal bypass valve to restore seal cooling.

D. All seal cooling to NCP 1C is lost. Secure NCP 1C to prevent further seal damage.

Question: 35
(1 point)

Unit 2 is in Mode 5 with alignment of the KC system for parallel operations per OP/1/A/6400/005 (Component Cooling System). Given the following conditions and events:

- 2A1, 2B1, and 2B2 KC Pumps are in service.
- Both 2ETA and 2ETB are aligned to Unit 1 offsite power
- An 86S relay actuates on 2ETB
- All systems respond appropriately in automatic.

Assuming no operator actions, which Unit 2 KC pumps are in service?

A. 2A1 KC pump only

B. 2A1 and 2A2 KC pumps only

C. 2A1, 2B1, and 2B2 KC pumps only

D. 2A1, 2A2, 2B1 and 2B2 KC pumps

Question: 36
(1 point)

Given the following sequence of events and conditions:

- A pressurizer PORV opens spuriously and will not close
- 3 minutes after the PORV opens, the block valve is closed.
- NC pressure is 1500 psig
- NC temperature is 550 °F
- PRT pressure is 45 psig

What is the approximate pressurizer PORV tailpipe temperature?

Reference provided

A. 270 °F

B. 290 °F

C. 310 °F

D. 320 °F

Question: 37
(1 point)

Given the following conditions and sequence of events:

- Unit 1 was operating at 100% power.
- The crew has entered AP/1/A/5500/016 (Malfunction of Nuclear Instrumentation System) due to N-42 lower detector failing LOW
- IAE has not yet placed the required bistables in the trip condition per AP/1/A/5500/016.
- A complete loss of 1ERPD occurs

What procedure takes priority for these conditions?

A. Continue in AP/1/A/5500/016

B. Enter AP/1/A/5500/029 (Loss of Vital or Aux Control Power)

C. Enter AP/1/A/5500/003 (Load Rejection)

D. Enter EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)

Question: 38
(1 point)

Which one of the following selections correctly matches the reactor trip signals to their limiting accident/protection?

	Reactor Trip Signal	Limiting Accident/Protection
A.	OPDT	DNB
	OTDT	Excessive fuel centerline temperature
	Pzr High Level	NC system integrity
	Pzr Low Pressure	DNB
B.	OPDT	Excessive fuel centerline temperature
	OTDT	DNB
	Pzr High Level	DNB
	Pzr Low Pressure	NC system integrity
C.	OPDT	Excessive fuel centerline temperature
	OTDT	DNB
	Pzr High Level	NC system integrity
	Pzr Low Pressure	DNB
D.	OPDT	NC System integrity
	OTDT	Excessive fuel centerline temperature
	Pzr High Level	DNB
	Pzr Low Pressure	DNB

Question: 39
(1 point)

Initial Conditions:

- Unit 1 was performing a heatup following a refueling outage
- NC Temperature was 400 °F
- NC pressure was 1600 psig
- "A" and "B" shutdown banks were withdrawn
- Containment Pressure Channel II failed high

Current Conditions:

- 1ERPD has lost power
- Containment pressure channels read:
  - Channel I: 0 psig
  - Channel II: +5 psig
  - Channel III: 0 psig
  - Channel IV: -5 psig

Which of the following statements explains the impact on the Engineered Safeguards Features (ESF) system and expected operator actions?

A. Only Train "A" safety injection actuates.
Implement AP/1/A/5500/005, Reactor Trip or Inadvertent S/I Below P-11.

B. Only Train "A" safety injection actuates.
Implement EP/1/A/5000/E-0, Reactor Trip or Safety Injection.

C. Train "A" and "B" safety injection actuates.
Implement AP/1/A/5500/005, Reactor Trip or Inadvertent S/I Below P-11.

D. Train "A" and "B" safety injection actuates.
Implement EP/1/A/5000/E-0, Reactor Trip or Safety Injection.

Question: 40
(1 point)

Which one of the following is the type of power supplied to the YV Chillers?

A. 600V unit power

B. 4160 V essential power

C. 4160 V blackout power

D. 6900 V unit power

Question: 41
(1 point)

Given the following:

- 1AD-13, D/8 "GLYCOL EXPANSION TNK LO-LO LVL" - LIT
- BOP notes that the Unit 1 NF containment isolation valves have closed

Where does the bypass valve for pressure relief between the isolation valves relieve to and from what location may the Glycol Expansion Tank Lo-Lo Level interlock be bypassed?

A. Glycol Expansion Tank / local NF control panel

B. Glycol Expansion Tank / main control room

C. Glycol Mixing and Storage Tank / local NF control panel

D. Glycol Mixing and Storage Tank / main control room

Question: 42
(1 point)

Given the following:

- A large break LOCA has occurred.
- Containment pressure is 3.2 psig and slowly decreasing.
- The crew has just transitioned to EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation)

What is the minimum containment sump level that will support operation of all ECCS pumps and the NS pumps?

A. 0.5 ft

B. 2.5 ft

C. 3.3 ft

D. 5.0 ft

Question: 43
(1 point)

Given the following sequence of events:

1200 Unit 1 reactor tripped from 100% power due to a large break LOCA

1236 FWST level is 36%
Containment pressure is 3.8 psig

1240 1NI-185A (ND Pump 1A Cont Sump Suct) is <u>not</u> open and efforts to open it from the control room have failed.

1241 1A ND pump is secured.

1245 NLOs have been dispatched to manually open 1NI-185A.

1300 NLOs report 1NI-185A is fully open.

1301 1A ND pump is started.

1305 FWST level is 16%
Containment pressure is 3.1 psig

Which one of the following describes the status of the 1A NS pump at 1245 and what is the earliest time that ND Aux Spray can be placed in service?

A. 1A NS pump was running; 1250

B. 1A NS pump was running; 1301

C. 1A NS pump was off; 1250

D. 1A NS pump was off; 1301

Question: 44
(1 point)

Given the following conditions and sequence of events:

- Unit 1 is manually tripped due to a loss of normal feedwater.
- NLOs have manually isolated CA flow to 1B S/G and level is noted to be 96% on NR level gauges.

Which of the following consequences have increased risk for 1B S/G based on the current water level in that S/G?

1. Failure of S/G PORV to actuate
2. Failure of SM safety valves to reseat following an actuation
3. Water hammer upon initiation of steam flow
4. Mechanical failure of the main steam lines

A. 1 and 2 only

B. 3 and 4 only

C. 1, 2 and 3

D. 2, 3 and 4

Question: 45
(1 point)

Unit 1 is at 75% power when a plant trip occurs due to P-14 actuation. Given the following events and conditions:

- The plant is currently stable.
- The steam dumps have just closed at no-load Tave.
- Steam generator NR levels are 35% in unaffected steam generators and 80% in the affected steam generator.

What action must the operator take to reset CF isolation?

A. Lower the affected steam generator level, cycle the reactor trip breakers and depress the CF isolation reset pushbuttons.

B. Lower the affected steam generator level and cycle the reactor trip breakers.

C. Cycle the reactor trip breakers and depress the CF isolation reset pushbuttons.

D. Cycle the reactor trip breakers only.

Question: 46
(1 point)

Given the following:

- Unit 2 was operating at 100% power.
- 2A steamline ruptured inside containment resulting in containment pressure rapidly increasing to 3.7 psig.
- Current containment pressure is 2.4 psig and slowly decreasing.
- The crew has just verified that total CA flow is greater than 450 gpm per step 18.a of EP/2/A/5000/E-0 (reactor Trip or Safety Injection).

Within what operating band should the BOP be attempting to control S/G N/R levels?

A. Between 11% and 50%

B. Between 29% and 50%

C. Between 9% and 62%

D. Between 21% and 62%

Question: 47
(1 point)

Given the following:

- 2B D/G automatically started due to the incoming breaker to 2ETB spuriously opening.
- While checking D/G operating parameters, the crew notes that D/G 2B "VOLTS" is 4300 V.
- At the direction of the CRS, the BOP adjusts voltage to normal.

How will D/G 2B output "AMPS" and "P/F" indications respond to this adjustment?

	AMPS	P/F
A.	increase	less lagging
B.	increase	stay the same
C.	decrease	less lagging
D.	decrease	stay the same

Question: 48
(1 point)

Which of the following receives power from 250VDC Auxiliary Power System?

A. D/G Fuel Oil Booster Pump

B. Reactor Trip Switchgear Control

C. Unit 1 Turbine Emergency Bearing Oil Pump

D. Power Operated Relief Valves Solenoids (both NC and SV systems)

Question: 49
(1 point)

Unit 1 was operating at 10% power preparing to roll the turbine. Given the following sequence of events:

0200 – 1A D/G Battery Charger 1DGCA fails.
0700 – D/G 1A Panel, E/5 "LOSS OF DC CONTROL POWER" - LIT
0900 - A tornado results in a complete loss of the switchyard.

Assuming no actions have been taken to address the failed charger, which one of the following statements correctly describes the operating status of the 1A D/G and the reason for this status?

A. The 1A D/G starts because the auto-start function is not dependent on DC control power.

B. The 1A D/G starts because the control power is supplied from vital power through auctioneering diode 1VADA.

C. The 1A D/G started but did not tie to the bus because the sequencer has lost all control power.

D. The 1A D/G did not start because it has lost all control power.

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Question: 50
(1 point)

Given the following conditions and sequence of events:

- Unit 2 was operating at 100% power when a LOCA occurred
- Containment pressure peaked at 2.6 psig and is slowly decreasing
- 2~~1~~A CA Pump failed to start
- "A" train ECCS and D/G load sequencer was reset
- 2~~1~~A CA Pump was manually started
- A complete loss of switchyard occurs

Assuming no operator actions since the loss of the switchyard, which of the following is a complete list of the ECCS pumps currently in service?

A. 2A NV, 2A NI, 2A ND, 2B NV, 2B NI, 2B ND

B. 2A NV, 2B NV, 2B NI, 2B ND

C. 2B NV, 2B NI, 2B ND

D. 2A NV, 2B NV

Question: 51
(1 point)

Unit 1 is operating at 100% power. A plant operator reports the following:

- D/G 1A Panel, B/8 "LOW VG AIR TANK PRESS" - LIT
- VG receivers starting air pressure is stable at 149 psig

Which one of the following statements correctly describes the state of readiness of the 1A D/G?

A. The D/G can be manually started and is capable of one or two starts.

B. The D/G can be automatically started and is capable of one or two starts.

C. The D/G can be manually or automatically started and is capable of five starts.

D. The D/G cannot be manually or automatically started until the VG receiver is repressurized.

Question: 52
(1 point)

Given the following:

- Unit 1 is operating at 8% power preparing to place the turbine online
- A VQ release is in progress
  1EMF-39L (CONTAINMENT GAS (LO RANGE)) detector fails causing a Trip 2 alarm
- 1RAD-1, A/2 "1EMF-39 CONTAINMENT GAS HI RAD" is LIT
- 1RAD-1, F/5 "CABINET 1-2 TROUBLE" is LIT

1. What is the status of the Unit 1 Containment Evacuation alarm?
2. What is/are the minimum action(s) required to reinitiate the air release from containment?

A. 1. The Containment Evacuation alarm has actuated.
   2. Bypass the failed EMF detector per OP/0/A/6500/080 (EMF RP86A Output Modules) and then RESET the safety signal per OP/1/B/6100/010X (Annunciator Response for Radiation Monitoring Panel 1RAD-1)

B. 1. The Containment Evacuation alarm has NOT actuated.
   2. Bypass the failed EMF detector per OP/0/A/6500/080 (EMF RP86A Output Modules) and then RESET the safety signal per OP/1/B/6100/010X (Annunciator Response for Radiation Monitoring Panel 1RAD-1)

C. 1. The Containment Evacuation alarm has actuated.
   2. RESET the safety signal only per OP/1/B/6100/010X per (Annunciator Response for Radiation Monitoring Panel 1RAD-1)

D. 1. The Containment Evacuation alarm has NOT actuated.
   2. RESET the safety signal only per OP/1/B/6100/010X per (Annunciator Response for Radiation Monitoring Panel 1RAD-1)

Question: 53
(1 point)

1A RN pump is normally powered from:

A. 4160V bus 1ETA

B. 4160V bus 1FTA

C. 6900V bus 1TA long side

D. 6900V bus 1TC long side

Question: 54
(1 point)

Unit 2 is in Mode 3 with charging and letdown in normal alignment.

What affect does a total loss of VI have on the NV system?

A. Charging flow increases; letdown flow increases

B. Charging flow increases; letdown flow decreases

C. Charging flow decreases; letdown flow increases

D. Charging flow decreases; letdown flow decreases

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Question: 55
(1 point)

Unit 1 is operating at 100% power with a routine containment air release in progress through 1VQ-10 (VQ Fans Disch To Unit Vent).

1. At what containment pressure will 1VQ-10 first receive a "CLOSE" signal?
2. What is the basis for closing 1VQ-10 at that pressure?

A. 1. -0.08 psig
   2. Non-compliance with technical specification on containment pressure

B. 1. -0.08 psig
   2. Unexpected opening of ice condenser inlet doors

C. 1. 0 psig
   2. Non-compliance with technical specification on containment pressure

D. 1. 0 psig
   2. Unexpected opening of ice condenser inlet doors

DELETED

Question: 56
(1 point)

Unit 1 was in Mode 3 with shutdown banks withdrawn in preparation for startup.
Given the following:

- 1TD short side incoming breaker trips
- 1TD tie breaker does not automatically close

Which MG set(s) has/have a power supply available and what is the current status of the shutdown banks?

A. Only 1A MG set; shutdown banks are inserted

B. Only 1A MG set; shutdown banks are withdrawn

C. 1A and 1B MG sets; shutdown banks are inserted

D. 1A and 1B MG sets; shutdown banks are withdrawn

Question: 57
(1 point)

Initial conditions at 1300:

- Unit 2 was at 50% power
- Pressurizer level was at program level
- 2NV-312A (Chrg Line Cont Isol) spuriously closed and could not be reopened
- Operators have taken the following actions per AP/2/A/5500/012 (Loss of Charging or Letdown), Case I (Loss of Charging):
  - Secured letdown
  - Total charging flow has been reduced to 32 gpm
- Excess letdown can <u>not</u> be established

At approximately what time will the pressurizer become inoperable per Tech Spec 3.4.9 (Pressurizer)?

<u>Reference provided</u>

A. 1434

B. 1608

C. 1651

D. 1825

Question: 58
(1 point)

Unit 1 was operating at 70% when 1C S/G MEDIAN SELECTED Wide Range (WR) Level output to the Digital Feedwater Control System (DFCS) fails low.

How will the DFCS respond to this event?

A. DFCS will switch 1C S/G CF reg valve and CF bypass reg valve to MANUAL.

B. DFCS will substitute another S/G's WR level input into "C" loop.

C. DFCS will generate a "DFCS TROUBLE" alarm only.

D. DFCS will reduce S/G 1C WR level to 50%.

Question: 59
(1 point)

Unit 1 was operating at 100% power when a loss of offsite power caused a reactor trip. The crew has verified natural circulation in ES-0.1 (Reactor Trip Response). Ten minutes later, the operator notes that the thermocouple input to both plasma displays is malfunctioning.

Which one of the following correctly describes a valid indication that natural circulation is continuing?

A. S/G pressures are decreasing and T_{cold} is at S/G saturation temperature.

B. S/G saturation temperatures are decreasing and REACTOR VESSEL UR LEVEL indication is greater than 100%.

C. S/G pressures are decreasing and REACTOR VESSEL D/P indication is greater than 100%.

D. S/G pressure is at saturation pressure for T_{cold} and REACTOR VESSEL D/P indication is greater than 100%.

Question: 60
(1 point)

Unit 1 was operating at 100% when a design basis LOCA occurred. Radiation monitoring teams at the site boundary report that Iodine 131 dose is 5 Rem.

Which one of the following statements correctly describes the condition of the VE filters that would result in the dose readings noted at the site boundary?

A. 1A VE train failed to start on the safety injection

B. The prefilter/demisters are saturated

C. The charcoal filters are saturated

D. The HEPA filters are saturated

Question: 61
(1 point)

Unit 1 is in Mode 5 following refueling. All S/Gs were drained and have just been refilled with condensate water per Chemistry request.

The following conditions existed during the filling operation and have been verified to be the current conditions:

Primary conditions:

- 1A ND Hx inlet temperature 185 °F
- 1B ND Hx inlet temperature 185 °F
- NC pressure 218 psig

Secondary conditions:

- S/G 1A CF inlet temperature 71 °F
- S/G 1B CF inlet temperature 72 °F
- S/G 1C CF inlet temperature 68 °F
- S/G 1D CF inlet temperature 71 °F
- All S/Gs pressures are 0 psig.

Based on the reported conditions, what is the action required by Selected License Commitments?

A. Increase 1C S/G secondary temperature to greater than 70 °F within 30 minutes.

B. Increase 1C S/G secondary temperature to greater than 70 °F within 1 hour.

C. Reduce NC pressure to less than or equal to 200 psig within 30 minutes.

D. Reduce NC pressure to less than or equal to 200 psig within 1 hour.

Question: 62
(1 point)

Unit 1 is operating at 100% power.

1. How is EHC Emergency Manual Mode selected?
2. How do the control valves respond to a manual runback under the above conditions?

A. 1. automatically
   2. the control valves will operate per the valve curves

B. 1. automatically
   2. the control valves will NOT operate per the valve curves

C. 1. manually
   2. the control valves will operate per the valve curves

D. 1. manually
   2. the control valves will NOT operate per the valve curves

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Question: 63
(1 point)

Which one of the following Shutdown Waste Gas Decay Tanks (SWGDTs) is maintained at a low pressure per the limits and precautions of OP/0/A/6500/003A (Gaseous Waste System (Normal Operations)) and what maximum pressure does it specify?

A. SWGDT A; less than 5 psig

B. SWGDT A; less than 30 psig

C. SWGDT B; less than 5 psig

D. SWGDT B; less than 30 psig

Question: 64
(1 point)

VI system pressure is 98 psig.

Which one of the following statements correctly describes the sequence and position of VI system valves in response to a loss of VI header pressure as pressure continues to decrease?

A. VS-78 (VS supply to VI) opens at 80 psig
VI-500 (VI supply to VS) opens at 76 psig

B. VS-78 (VS supply to VI) closes at 80 psig
VI-500 (VI supply to VS) opens at 76 psig

C. VI-500 (VI supply to VS) closes at 80 psig
VS-78 (VS supply to VI) opens at 76 psig

D. VI-500 (VI supply to VS) closes at 80 psig
VS-78 (VS supply to VI) closes at 76 psig

Question: 65
(1 point)

Given the following conditions and sequence of events:

- 2A D/G auto-started due to a blackout on 2ETA
- The control room crew notes all loads were sequenced on as required
- A fuel oil line leak occurs resulting in a major fire in the 2A D/G room

Assuming no operator actions since the D/G auto-started:

1. How long will it take for the Cardox system to discharge once the fire is detected?
2. What is the status of the 2A D/G emergency ventilation after the Cardox system discharges?

A. 1. 6.5 minutes
   2. Running due to sequencer actuation

B. 1. 6.5 minutes
   2. Secured due to Cardox actuation

C. 1. 1.5 minutes
   2. Running due to sequencer actuation

D. 1. 1.5 minutes
   2. Secured due to Cardox actuation

Question: 66
(1 point)

During a control board walkdown, the crew notes that over the last 10 minutes turbine load has decreased from 1209 MW to 1207 MW while reactor power has increased from 99.87% to 100.05%. They suspect a steam leak.

Which set of the following indications could be used to confirm their suspicions?

1. % Steam flow
2. Steam pressure
3. Containment pressure
4. Containment humidity

A. 1, 2, 3

B. 1, 2, 4

C. 1, 3, 4

D. 2, 3, 4

Question: 67
(1 point)

Terrorists have broken through the security fence and set both Unit 1 main transformers on fire. Security has notified the operating crew that several terrorists are enroute to the control room.

What instructions are provided to the NLO dispatched to the 1ETA switchgear room and which procedure provides that guidance?

A. Perform a partial transfer to the SSF per AP/1/A/5500/017 (Loss of Control Room)

B. Transfer control to the SSF per AP/1/A/5500/017 (Loss of Control Room)

C. Perform a partial transfer to the SSF per AP/0/A/5500/045 (Plant Fire)

D. Transfer control to the SSF per AP/0/A/5500/045 (Plant Fire)

Question: 68
(1 point)

During a power increase to 100% power per OP/1/A/6100/003 (Controlling Procedure for Unit Operation), the "C" Heater Drain Pumps are placed in service at a minimum power level of _____. The purpose of this is to prevent the potential for _____.

A. 50% / excessive main feedwater pump discharge pressure

B. 70% / excessive main feedwater pump discharge pressure

C. 50% / deadheading of hotwell and booster pumps

D. 70% / deadheading of hotwell and booster pumps

Question: 69
(1 point)

Unit 1 is at 4% power, conducting a plant startup. Given the following events and conditions:

- One control bank "A" rod drops fully into the core
- NCS temperature decreases to 550°F

Which one of the following statements correctly describes an action that is required within 30 minutes by Technical Specifications?

A. Be in mode 2 with K_{eff} less than 1.0.

B. Restore rod group within alignment limits.

C. Verify shutdown margins within the limits specified in the COLR.

D. Adjust power range N/Is to increase reactor power so that reactor power and thermal power best estimate are equal.

Question: 70
(1 point)

A Unit 1 containment purge is in progress using OP/1/A/6450/015. Given the following events and conditions:

- 1EMF-39(L) (CONTAINMENT GAS (LO RANGE)) spiked to a Trip 2 condition then cleared

Which one of the following statements correctly describes the action required?

A. The VP release may not be reinitiated until RP draws a new containment air activity sample.

B. The VP release may be reinitiated after the spike clears. If 1EMF-39 spikes a second time, the release may also be reinitiated.

C. The VP release may be reinitiated after the spike clears. If 1EMF-39 spikes a second time, the release cannot be reinitiated without RP sampling containment air for activity.

D. The VP release may be reinitiated if grab samples are taken of Unit Vent activity during subsequent reinitiation.

Question: 71
(1 point)

While performing a valve lineup in the boric acid mixing room, an air line failure caused a severe airborne beta contamination problem. A worker received both internal and external contamination that was detected upon attempting to exit the RCA.

Which one of the exposures would exceed the 10CFR20 limit for the worker's annual shallow dose equivalent (SDE) exposure?

A. 55 Rem external dose to the lens of the eye.

B. 55 Rem external dose to the leg below the knee.

C. 17 Rem internal dose equivalent to the lens of the eye.

D. 17 Rem internal dose to the right forearm.

Question: 72
(1 point)

A radiation worker is repairing a valve in a contaminated area, which has the following radiological characteristics:

- The worker's present exposure is 1938 mrem for the year
- The RWP states:
  - General area dose rate = 30 mrem/hr
  - Airborne contamination concentration = 10.0 DAC

The job will take 2 hours if the worker wears a full-face respirator. It will only take 1 hour if the worker does not wear the respirator.

If the RP Manager grants all applicable dose extensions, which one of the following choices for completing this job would maintain the worker's exposure within the station administrative requirements?

A. The worker should not wear the respirator.
The dose received wearing a respirator will exceed site annual personnel dose limits.

B. The worker should not wear the respirator.
The calculated TEDE dose received will be less than if he does wear one.

C. The worker should wear the respirator.
The calculated TEDE dose received will be less than if he does not wear one.

D. The worker should wear the respirator.
He could exceed DAC limits.

Question: 73
(1 point)

The crew is responding to a spurious safety injection. Given the following validated CSF status tree indications:

- Subcriticality – GREEN
- Core Cooling – GREEN
- Heat Sink – GREEN
- NC Integrity – GREEN
- Containment – GREEN
- NC Inventory - YELLOW

Per OMP 1-7 (Emergency/Abnormal Procedure Implementation Guidelines):

1. Which control room crew position, by title, has primary responsibility for monitoring Critical Safety Function (CSF) status trees during EOP usage?
2. Based on current conditions how frequent should CSF status trees be monitored?

A. 1. OSM
   2. monitor every 10-20 minutes

B. 1. OSM
   2. monitor continuously

C. 1. STA
   2. monitor every 10-20 minutes

D. 1. STA
   2. monitor continuously

Question: 74
(1 point)

Which one of the following sets of critical safety functions (CSFs):

- is listed in the correct order per the CSF status trees from highest to lowest priority

  AND

- forms the bases for protection of the fuel and fuel cladding?

A.	1. Heat Sink	2. Core Cooling	3. Integrity
B.	1. Core Cooling	2. Heat Sink	3. NC Inventory
C.	1. Heat Sink	2. Subcriticality	3. NC Inventory
D.	1. Subcriticality	2. Heat Sink	3. Integrity

Question: 75
(1 point)

An offsite release is occurring due to a stuck open S/G PORV on 2C S/G which has a significant tube leak.

Which one of the following states:

1. The emergency facility that assumes responsibility for communications with offsite agencies including the NRC once it is activated?
2. What is the lowest classification level that requires this facility's activation?

A. 1. Technical Support Center (TSC)
   2. Alert

B. 1. Technical Support Center (TSC)
   2. Unusual Event

C. 1. Operations Support Center (OSC)
   2. Alert

D. 1. Operations Support Center (OSC)
   2. Unusual Event

Question: 76
(1 point)

Given the following Unit 1 conditions and sequence of events:

- NC system temperature is 208 °F
- NC system pressure is 350 psig
- 1A NV pump is red tagged to replace its 1ETA breaker
- 1B NI pump is white tagged
- 1A ND and 1B ND loops operating in residual heat removal mode
- An ND pump suction relief has spuriously lifted and has <u>not</u> reseated
- Both ND pumps have been secured per AP/1/A/5500/027 (Shutdown LOCA)

1. What is the correct procedure flowpath for this situation?
2. What is the limiting component that the current ECCS pump configuration is designed to protect from over-pressurization?

A. 1. Remain in AP/1/A/5500/027 (Shutdown LOCA)
   2. NC loop crossover pipe

B. 1. Transition to AP/1/A/5500/019 (Loss of Residual Heat Removal System)
   2. NC loop crossover pipe

C. 1. Remain in AP/1/A/5500/027 (Shutdown LOCA)
   2. Reactor vessel

D. 1. Transition to AP/1/A/5500/019 (Loss of Residual Heat Removal System)
   2. Reactor vessel

Question: 77
(1 point)

Unit 2 is at 3% power. Given the following sequence of events:

12/01/08	1100	2A NI pump tagged to replace the motor cooler.
12/03/08	0500	2B D/G tripped on high vibration during performance of PT/2/A/4350/002B (Diesel Generator 2B Operability Test).
12/03/08	0700	You complete turnover and take the position of CRS.

1. What is the <u>latest</u> time that entry into Mode 3 is <u>required</u> per Technical Specifications assuming both components remain inoperable?
2. When you take shift duty at 0700, can the ECCS design criteria for a large break LOCA be assumed to be met?

<u>Reference provided</u>

A. 1. 12/03/08 1200
   2. Yes

B. 1. 12/03/08 1200
   2. No

C. 1. 12/03/08 1600
   2. Yes

D. 1. 12/03/08 1600
   2. No

Question: 78
(1 point)

Unit 2 is at 100% power when an NLO reports the breaker for 2KC-56A (KC To ND Hx 2A Sup Isol) looks damaged. Upon investigation, the SPOC crew determines that 2KC-56A (KC To ND Hx 2A Sup Isol) will not open.

What is the minimum flow required through this valve when aligned for cold leg recirculation and for the situation above, what system is required to be declared inoperable?

A. 5000 gpm / 2A Train of KC

B. 5000 gpm / 2A Train of ND

C. 5700 gpm / 2A Train of ND

D. 5700 gpm / 2A Train of KC

Question: 79
(1 point)

Unit 1 is at 12% power following a refueling outage. Given the following conditions and sequence of events:

1200 1NCP5880 (NC Loop 1B Cold Leg Temp) failed low

1205 Unit 1 separated from the grid; the main turbine is carrying all in-house loads

1210 The crew has tripped the reactor, safety injected and entered EP/1/A/5000/E-0 (Reactor Trip or Safety Injection) based on the following indications:
- Charging flow is 125 gpm with letdown isolated
- PZR level is decreasing as a rate of 0.5% /minute

1213 1EDA loses all power due to a fault

1220 The crew is preparing to kick out of EP/1/A/5000/E-0 and notes the following indications:
- Containment pressure is stable at 0.08 psig
- All S/G pressures are stable at 1100 psig
- 1EMF-33 (Condenser Air Ejector Exhaust) Trip 2 is LIT
- Off-normal Critical Safety Function status as follows:
  - Containment is MAGENTA
  - Core Cooling is ORANGE
  - Heat Sink is YELLOW
  - NC Integrity is RED
  - NC Inventory is YELLOW

What is the next procedure to be entered?

A. Enter EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant)

B. Enter EP/1/A/5000/E-3 (Steam Generator Tube Rupture)

C. Enter EP/1/A/5000/FR-C.2 (Response to Degraded Core Cooling)

D. Enter EP/1/A/5000/FR-P.1 (Response to Imminent Pressurized Thermal Shock)

Question: 80
(1 point)

Given the following:

- Unit 2 was operating at 100%.
- At 1000, charger 2ECA output breaker opened due to an overvoltage condition.
- The 2EDA tie breaker to 2EDC can not be closed.
- At 1130, battery 2EBA voltage dropped below the voltage required per Technical Specifications.

Which one of the following describes the latest time that bus 2EDA can be restored to prevent entering a shutdown action and which procedure will be entered initially to respond to this failure?

A. 1200; EP/2/A/5500/E-0 (Reactor Trip or Safety Injection)

B. 1200; AP/2/A/5500/029 (Loss of Vital or Aux Control Power)

C. 1330; EP/2/A/5500/E-0 (Reactor Trip or Safety Injection)

D. 1330; AP/2/A/5500/029 (Loss of Vital or Aux Control Power)

Question: 81
(1 point)

Unit 1 is operating at 100% power.
Unit 2 is in a refueling outage with 2EMXH aligned to Unit 1 power.

Given the following conditions and sequence of events:

0530 1AD-11, G/6 "SWGR 1ETA DEGRADED BUS VOLTAGE" is LIT
1AD-11, H/6 "SWGR 1ETB DEGRADED BUS VOLTAGE" is LIT
1AD-11, K/6 "230KV SWITCHYARD VOLTAGE LO" is LIT

0535 The STA notes by OAC trends that 1ETA and 1ETB minimum voltages were 3620V and 3637V respectively and are now increasing.

1. At 0530, what is the earliest time required for Unit 1 to enter Mode 3 per Technical Specifications?
2. At 0535, assuming no operator actions, what is the status of D/G 1A and D/G 1B?

A. 7 hours due to TS 3.0.3; both running

B. 7 hours due to TS 3.0.3; both secured

C. 6 hours due to TS 3.7.5 (Auxiliary Feedwater (AFW) System); both running

D. 6 hours due to TS 3.7.5 (Auxiliary Feedwater (AFW) System); both secured

---

Question: 82
(1 point)

Unit 1 is in Mode 1.

1. With a Boric Acid Tank (BAT) temperature of 63°F, what is the most limiting required Technical Specification/Selected License Commitment action time?
2. What plant event requires emergency boration using 1NV-236B (Boric Acid to NV Pumps Suct)?

A. 1. 1 hour
   2. In response to an ATWS

B. 1. 72 hours
   2. In response to an ATWS

C. 1. 1 hour
   2. When control rods are below the Lo-Lo insertion limits

D. 1. 72 hours
   2. When control rods are below the Lo-Lo insertion limits

Question: 83
(1 point)

Unit 1 is operating at 100% power. Unit 2 is in No Mode. The control room has become uninhabitable due to chlorine gas intrusion and control has been shifted to the Auxiliary Shutdown Complex per AP/1/A/5500/017 (Loss of Control Room).

1. How is adequate primary side inventory assured?
2. For the situation above, which one of the following sets of valves would require a temporary modification to prevent them from automatically aligning should a safety injection occur?

A.
1. Automatic swap of NV pump suctions to the FWST
2. 1NI-9A (NV Pmp C/L Inj Isol) and 1NI-10B (NV Pmp C/L Inj Isol)

B.
1. Automatic swap of NV pump suctions to the FWST
2. 1ND-26 (ND Hx 1A Outlet Ctrl) and 1ND-60 (ND Hx 1B Outlet Ctrl)

C.
1. Manual swap of NV pump suctions to the FWST
2. 1NI-9A (NV Pmp C/L Inj Isol) and 1NI-10B (NV Pmp C/L Inj Isol)

D.
1. Manual swap of NV pump suctions to the FWST
2. 1ND-26 (ND Hx 1A Outlet Ctrl) and 1ND-60 (ND Hx 1B Outlet Ctrl)

Question: 84
(1 point)

Assuming no additional actions, which one of the following situations will result in a required Technical Specification shutdown within the next 30 days?

A. 1VI-77B (VI Cont Isol) fails in an intermediate position

B. Both lower personnel airlock doors closed and locked with both seals deflated on the outer door only

C. Both upper personnel airlock doors closed and locked with the airlock door interlock mechanism inoperable

D. 1VQ-15B (Cont Air Add Cont Isol) fails in an intermediate position and 1VQ-16A (Cont Air Add Cont Isol) is closed and de-activated

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Question: 85
(1 point)

Regarding the use of EP/1/A/5000/FR-Z.3 (Response To High Containment Radiation):

1. At what minimum reading on 1EMF 53A (Containment High Range) is the YELLOW path for Containment High Radiation valid?
2. What mitigative strategy does this procedure direct to reduce activity in the containment atmosphere?

A. 1. 35 R/hr
   2. Start Containment Auxiliary Charcoal Filter Units.

B. 1. 15 R/hr
   2. Start Containment Auxiliary Charcoal Filter Units.

C. 1. 35 R/hr
   2. Ensure the VE system is in service and vent containment to the annulus using the VY system.

D. 1. 15 R/hr
   2. Ensure the VE system is in service and vent containment to the annulus using the VY system.

Question: 86
(1 point)

Unit 2 is in preparations for startup with the shutdown banks withdrawn and the control banks inserted. Given the following:

- 2AD-7 C/1 NCP #1 "SEAL LEAKOFF HI FLOW" is LIT
- 2B NCP seal leakoff is 6.5 gpm
- 2B NCP Seal Outlet temperature is slowly increasing
- The crew enters AP/2/A/5500/008 (Malfunction of Reactor Coolant Pump)

What is the maximum time 2B NCP can remain in service and what procedure does AP/2/A/5500/008 direct the crew to enter once the pump is tripped?

A. 5 minutes; EP/2/A/5000/E-0 (Reactor Trip or Safety Injection)

B. 5 minutes; AP/2/A/5500/004 (Loss of Reactor Coolant Pump)

C. 8 hours; OP/2/A/6100/002 (Controlling Procedure For Unit Shutdown)

D. 8 hours; AP/2/A/5500/004 (Loss of Reactor Coolant Pump)

Question: 87
(1 point)

The night shift surveillance readings for Lake Wylie temperature over the past several days are as follows:

- 8/01/08 – 87.50° F.
- 8/02/08 – 88.25° F.
- 8/03/08 – 89.00° F.
- 8/04/08 – 89.75° F.
- 8/05/08 – 90.50° F.

1. Assuming lake temperature continues to increase at a constant rate, on what date will Lake Wylie temperature <u>first</u> exceed the requirements of SLC 16.9-14 (Lake Wylie Water Temperature)?
2. What affect, if any, will this higher lake temperature have on the ability of the NS system to affect containment pressure following a large break LOCA?

A. 1. 8/09/08
   2. Minimal impact prior to ice melt, but significant impact later in the accident sequence when the ice has been depleted.

B. 1. 8/09/08
   2. Minimal impact during the entire accident sequence since lake temperature is still below the design basis accident assumptions.

C. 1. 8/12/08
   2. Minimal affect prior to ice melt, but significant affect later in the accident sequence when the ice has been depleted.

D. 1. 8/12/08
   2. Minimal impact during the entire accident sequence since lake temperature is still below the design basis accident assumptions.

Deleted

Question: 88
(1 point)

Given the following:

- Rod control is in MANUAL.
- Turbine power has decreased from 1227 MW to 1214 MW and stabilized.
- The crew has just entered AP/1/A/5500/028 (Secondary Steam Leak).

What single steam relief valve passing 20% of its full flow would produce the conditions noted and what actions will be directed per AP/1/A/5500/028 based on the above conditions?

A. A steam line safety; trip the reactor and go to EP/1/A/5000/E-0 (Reactor Trip or Safety Injection.

B. A S/G PORV; trip the reactor and go to EP/1/A/5000/E-0 (Reactor Trip or Safety Injection.

C. A S/G PORV; initiate a unit shutdown per AP/1/A/5500/009 (Rapid Downpower)

D. A steam line safety; initiate a unit shutdown per AP/1/A/5500/009 (Rapid Downpower)

Question: 89
(1 point)

Unit 1 is operating at 100% power. Given the following:

- 1A D/G was manually started by NLOs for monthly surveillance testing
- A grid instability and relay failures caused all Unit 1 Switchyard PCBs to open
- 1B D/G failed to start
- Annunciator D/G 1A Panel, A/4 "TRIP LOW PRESS LUBE OIL" – LIT
- The ensuing transient resulted in a 1B S/G tube rupture

Which procedure will be used to isolate the ruptured S/G in this situation, and what procedural guidance is given regarding isolation of the ruptured steam generator?

A. EP/1/A/5000/E-3 (Steam Generator Tube Rupture) is used to isolate the ruptured S/G as soon as it is identified.

B. EP/1/A/5000/E-3 (Steam Generator Tube Rupture) is used to isolate the ruptured S/G only if S/G NR level is greater than 11%.

C. EP/1/A/5000/ECA-0.0 (Loss of All AC Power) is used to isolate the ruptured S/G as soon as it is identified.

D. EP/1/A/5000/ECA-0.0 (Loss of All AC Power) is used to isolate the ruptured S/G only if S/G NR level is greater than 11%.

Question: 90
(1 point)

Both units were operating at 100% power with 1A RN pump in service.

1A D/G was operating in parallel for surveillance testing when the following conditions and sequence of events occurred:

- 1AD-12, A/2 "RN ESSENTIAL HDR A PRESSURE – LO" - LIT
- 2AD-12, A/2 "RN ESSENTIAL HDR A PRESSURE – LO" - LIT
- 1AD-12, A/5 "RN ESSENTIAL HDR B PRESSURE – LO" - LIT
- 2AD-12, A/5 "RN ESSENTIAL HDR B PRESSURE – LO" - LIT
- NLO reported that he evacuated the 1A D/G room due to flooding.
- 1A D/G was immediately secured by the control room crew.
- All annunciators listed above continue to remain LIT.
- The crew entered and took all actions per AP/0/A/5500/030 (Plant Flooding) necessary to stop the flooding.

1. At what RN header pressure do the annunciators <u>first</u> come into alarm?
2. What is the current overall status related to Tech Spec 3.7.8 (Nuclear Service Water System (NSWS))?

A. 1. 40 psig decreasing
   2. Unit 1 in a 72 hour action, Unit 2 operable

B. 1. 40 psig decreasing
   2. Both units in a 72 hour action

C. 1. 46 psig decreasing
   2. Unit 1 in a 72 hour action, Unit 2 operable

D. 1. 46 psig decreasing
   2. Both units in a 72 hour action

Question: 91
(1 point)

Unit 2 is in Mode 6 performing core unloading when Spent Fuel Pool level is noted at 22 feet above the fuel assemblies.

1. Which one of the following is a required action for the above condition per Technical Specifications?
2. What is the basis for maintaining a minimum acceptable water level?

A.
1. Immediately suspend movement of irradiated fuel assemblies
2. Ensures shielding during fuel movement and to meet the assumptions for iodine decontamination factors following a fuel handling accident

B.
1. Immediately suspend movement of irradiated fuel assemblies
2. Ensures that there is a sufficient volume of water above the fuel assemblies to provide backup decay heat removal

C.
1. Within 1 hour, initiate action to restore spent fuel pool level to within limits.
2. Ensures shielding during fuel movement and to meet the assumptions for iodine decontamination factors following a fuel handling accident

D.
1. Within 1 hour, initiate action to restore spent fuel pool level to within limits.
2. Ensures that there is a sufficient volume of water above the fuel assemblies to provide backup decay heat removal

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Question: 92
(1 point)

Given the following:

- Unit 2 has experienced a Safety Injection.
- All S/G pressures are 1000 psig and stable.
- The crew has entered EP/2/A/5000/E-3 (Steam Generator Tube Rupture) due to 2EMF-10 (Steamline A) Trip 1 light being LIT.
- The BOP informs the OSM that 2RAD-3, F/3 (CABINET TROUBLE) is LIT
- The OSM believes the EMF detector may have failed.

1. What method can the crew use to determine the validity of the EMF indication?
2. Once the indication is determined to be false, which of the following describes the correct procedure transition?

A.
1. Verify Trip 1 alarm on adjacent steamline EMF (2EMF-13 (Steamline D)) is DARK
2. Transition to EP/2/A/5000/ES-0.0 (Rediagnosis)

B.
1. Verify Trip 1 alarm on adjacent steamline EMF (2EMF-13 (Steamline D)) is DARK
2. Evaluate tape flags in EP/2/A/5000/E-3 and then transition to EP/2/A/5000/E-1 (Loss of Reactor or Secondary Coolant)

C.
1. Request that RP frisk cation columns
2. Transition to EP/2/A/5000/ES-0.0 (Rediagnosis)

D.
1. Request that RP frisk cation columns
2. Evaluate tape flags in EP/2/A/5000/E-3 and then transition to EP/2/A/5000/E-1 (Loss of Reactor or Secondary Coolant)

Question: 93
(1 point)

Given the following:

- Unit 1 is at 100% power.
- Unit 2 is in No Mode.
- NLOs were running the SSF D/G per PT/0/A/4200/017 (Standby Shutdown Facility Diesel Test) when a fuel oil leak resulted in a fire.
- The SSF sprinkler system failed to actuate which resulted in damage to the SSF D/G.
- The Plant Fire Brigade extinguished the fire 20 minutes later.

What is the current emergency classification and what procedure will be used to address this situation?

Reference provided

A. Unusual Event; AP/1/A/5500/017 (Loss of Control Room) Case 2, "Loss of Plant Control Due to Fire or Security Event"

B. Unusual Event; AP/0/A/5500/045 (Plant Fire)

C. Alert; AP/1/A/5500/017 (Loss of Control Room) Case 2, "Loss of Plant Control Due to Fire or Security Event"

D. Alert; AP/0/A/5500/045 (Plant Fire)

Question: 94
(1 point)

Unit 1 is at 100% with 1A CA pump tagged for preventative maintenance. Given the following conditions and sequence of events:

- The main turbine trips due to faulty MSR high level signal
- NLOs were dispatched and opened the reactor trip breakers locally
- CAPT tripped on overspeed
- 1B CA Pump is found to have no indicating lights and no discharge pressure or flow indicated
- NLO reports 1B CA Pump control power is unavailable
- CAPT was successfully reset and restarted

- Current S/G parameters are:

	1A	1B	1C	1D
N/R level	10%	7%	9%	10%
CA flow	105 gpm	105 gpm	115 gpm	110 gpm

Which one of the following is the correct Emergency Action Level and the first required notification to plant personnel for this current conditions?

Reference provided

A. Enter a General Emergency and notify all plant personnel to perform a site assembly

B. Enter a General Emergency and notify non-essential plant personnel to perform a site evacuation

C. Enter a Site Area Emergency and notify all plant personnel to perform a site assembly

D. Enter a Site Area Emergency and notify non-essential plant personnel to perform a site evacuation

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Question: 95
(1 point)

Given the following:

- Core reload is in progress with 1A ND train in service.
- 1B ND train is inoperable.
- The fuel handling SRO requests 1A ND train be secured to allow a fuel assembly to be placed near the cold leg nozzle.

1. What is the maximum time 1A ND train can remain shutdown per Technical Specification 3.9.4 (Residual Heat Removal (RHR) and Coolant Circulation - High Water Level)
2. Why is boron concentration of any NC System make-up strictly limited with all ND loops shutdown?

A. 1. 30 minutes
   2. Lack of adequate NC System temperature monitoring

B. 1. 30 minutes
   2. Lack of adequate mixing of NC System water

C. 1. 1 hour
   2. Lack of adequate NC System temperature monitoring

D. 1. 1 hour
   2. Lack of adequate mixing of NC System water

Question: 96
(1 point)

Unit 1 is operating in Mode 3 preparing for a reactor startup following a refueling outage. Given the following events and conditions:

- NC Pump 1C is running.
- Reactor trip breakers are tagged open.
- Maintenance determines that the MOV test data from the outage indicates that the torque switches for 1ND-65B (ND TRAIN 1B HOT LEG INJ ISOL) have been set too low.
- The SWM requests OSM approval to tag closed 1ND-65B for repairs.

Which one of the following statements correctly describes the operating restrictions and implications of tagging closed 1ND-65B?

A. 1ND-65B may be tagged closed for 72 hours if the steam generator in the running NC loop is operable.

B. 1ND-65B may <u>not</u> be tagged closed because this would make both trains of ND inoperable.

C. 1ND-65B may <u>not</u> be tagged closed unless two NCPs are running with operable steam generators.

D. 1ND-65B may be tagged closed if 1ND-65B is restored to operation prior to transitioning to mode 2.

Question: 97
(1 point)

Which unit has a lower setpoint for P-14, and what is the basis for limiting maximum water level in the S/Gs?

A. Unit 1 / Limit energy release into containment following a steam line break

B. Unit 2 / Limit energy release into containment following a steam line break

C. Unit 1 / Maintain offsite dose within assumed limits following a SGTR

D. Unit 2 / Maintain offsite dose within assumed limits following a SGTR

Question: 98
(1 point)

Unit 2 is operating at 100% power. Maintenance has requested entry into the lower airlock. The work will require propping open the airlock vestibule door (CAD door) and the outer airlock door. The inner airlock door will remain closed.

For this situation, per Site Directive 3.1.2 (Access to Reactor Building And Areas Having High Pressure Steam Relief Devices) whose permission is required to issue the access keys to this area and what is inoperable based on Technical Specifications?

A. WCC SRO and Radiation Protection; the Annulus Ventilation System

B. WCC SRO only; the Annulus Ventilation System

C. WCC SRO and Radiation Protection; the Reactor Building

D. WCC SRO only; the Reactor Building

Question: 99
(1 point)

A contract worker is performing a task in an area with 5000 dpm/100 cm^2 (beta, gamma) contamination. His coworkers have reported he is acting erratically and believe he is "on" something. While waiting for supervision and security to arrive the individual falls and is injured. The individual is contaminated and must be transported offsite for medical treatment.

What is the correct posting for the work area and what is the <u>first</u> <u>required</u> NRC notification time for this event?

A. Contaminated Area; 24 hours

B. Highly Contaminated Area; 24 hours

C. Contaminated Area; 8 hours

D. Highly Contaminated Area; 8 hours

Question: 100
(1 point)

Given the following conditions:

- Unit 2 has experienced a small break LOCA
- The crew has transitioned to EP/2/A/5000/ES-1.2 (Post LOCA Cooldown and Depressurization)
- Containment pressure is 4.5 psig and decreasing slowly
- Present pressure indications are:
  - PZR PRESS Channel 1 - 1815 psig
  - PZR PRESS Channel 2 - 1795 psig
  - PZR PRESS Channel 3 - Failed High
  - PZR PRESS Channel 4 - Failed High
  - LOOP B HOT LEG W/R PRESS - 1920 psig
  - LOOP C HOT LEG W/R PRESS - Failed Low

1. Which instrument(s) above will provide the most reliable indication of current primary system pressure?
2. Based on the indications provided, is the LCO for Technical Specification 3.3.3 (PAM Instrumentation) met?

A. 1. LOOP B HOT LEG W/R PRESS
   2. No

B. 1. PZR PRESS Channels 1 and 2
   2. No

C. 1. LOOP B HOT LEG W/R PRESS
   2. Yes

D. 1. PZR PRESS Channels 1 and 2
   2. Yes

Reference List for: 2008 SRO NRC Examination

Databook Figure 43 (Generator Capability Curve)

EP/1/A/5000/ECA-1.1 (Step 19)

EP/1/A/5000/ECA-1.1 (Enclosure 5)

ASME Steam Tables

Pressurizer volume (gal) to level (%) graph

Technical Specification 3.5.2

Technical Specification 3.8.1

RP/0/A/5000/001 Classification of Emergency

Source: CNM-1300.00153, Fig. 17-3, Dwg 435HA148 (Rev. 2)

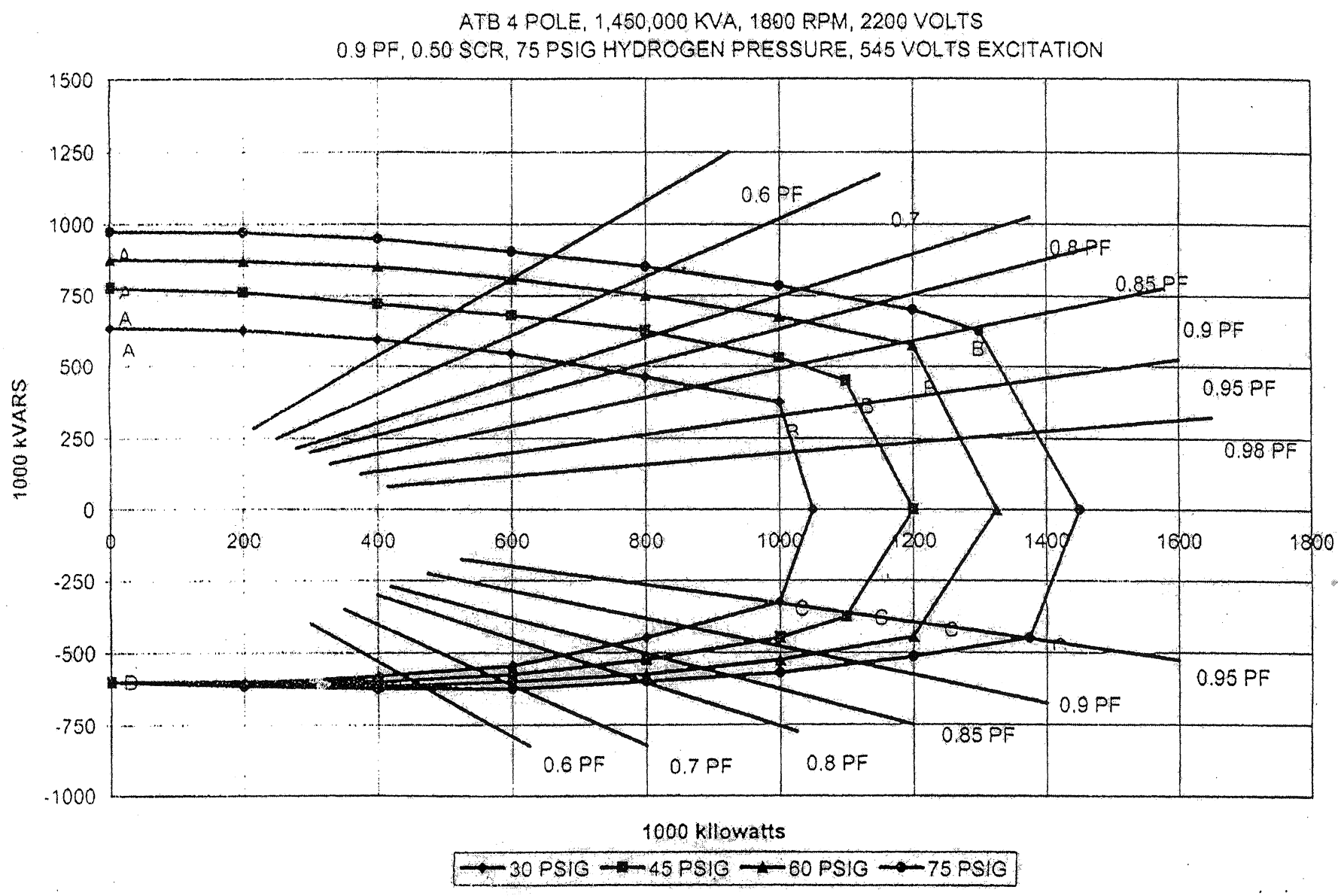


Figure 43 - Generator Capability Curves

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

19. **Verify S/I termination criteria as follows:**

a. Verify RVLIS indication is adequate as follows:

__ • **IF** all NC pumps are off, **THEN** verify "REACTOR VESSEL LR LEVEL" - GREATER THAN 61%.

__ • **IF** at least one NC pump is on, **THEN** verify "REACTOR VESSEL D/P" - GREATER THAN REQUIRED D/P FROM TABLE BELOW:

__ a. **GO TO** Step 26.

Number of NC Pumps On	Required "REACTOR VESSEL D/P"			
	TRN A With NC Pump 1A		TRN B With NC Pump 1C	
	On	Off	On	Off
4	80%	N/A	80%	N/A
3	60%	32%	60%	32%
2	45%	20%	45%	20%
1	35%	14%	35%	14%

__ b. NC subcooling based on core exit T/Cs - GREATER THAN 50°F.

b. Perform the following:

__ 1) Determine minimum S/I flow required. **REFER TO** Enclosure 5 (Minimum S/I Flowrate Versus Time After Trip).

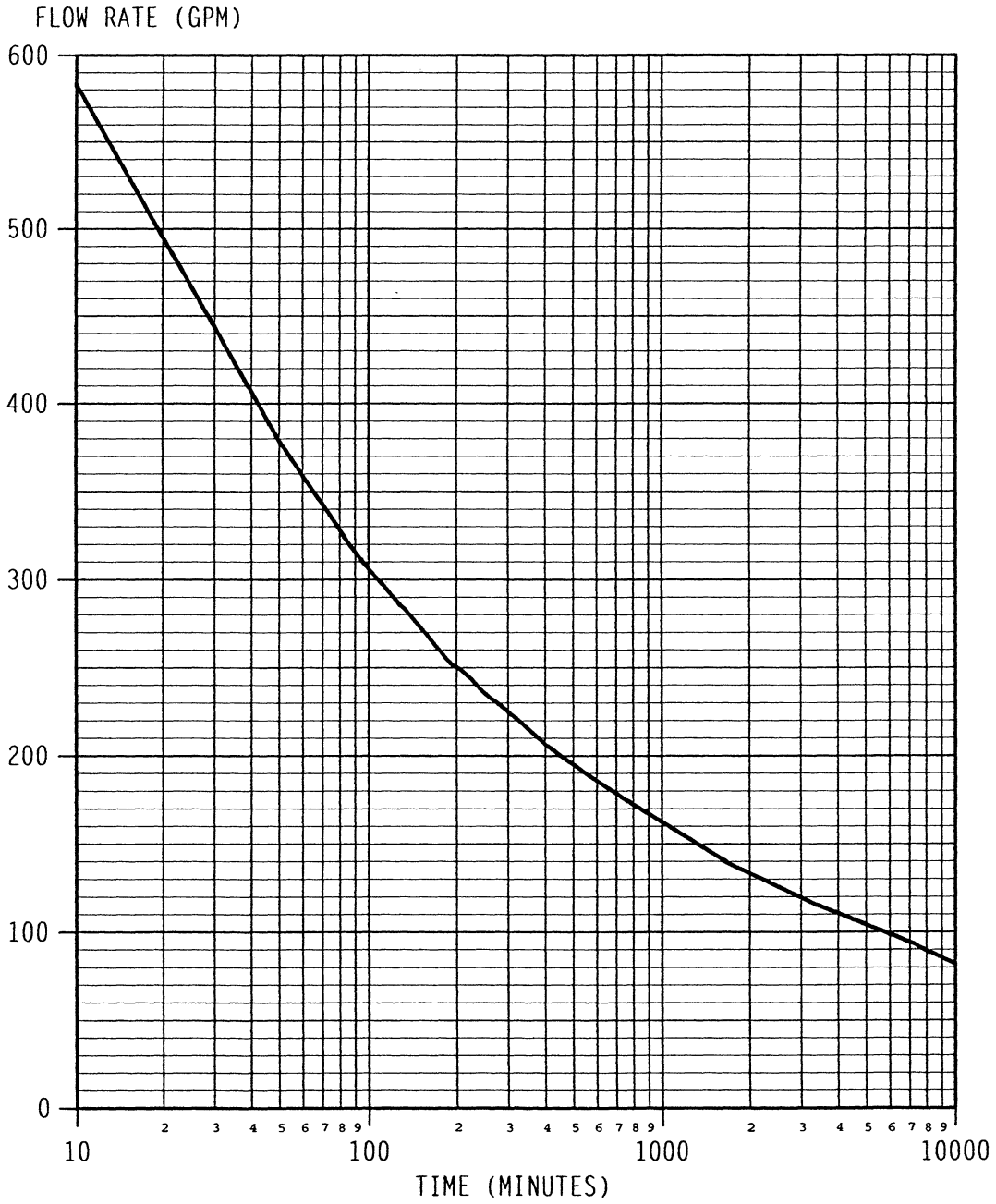
2) Stop S/I pumps as required to obtain the following:

__ • Minimize S/I flow

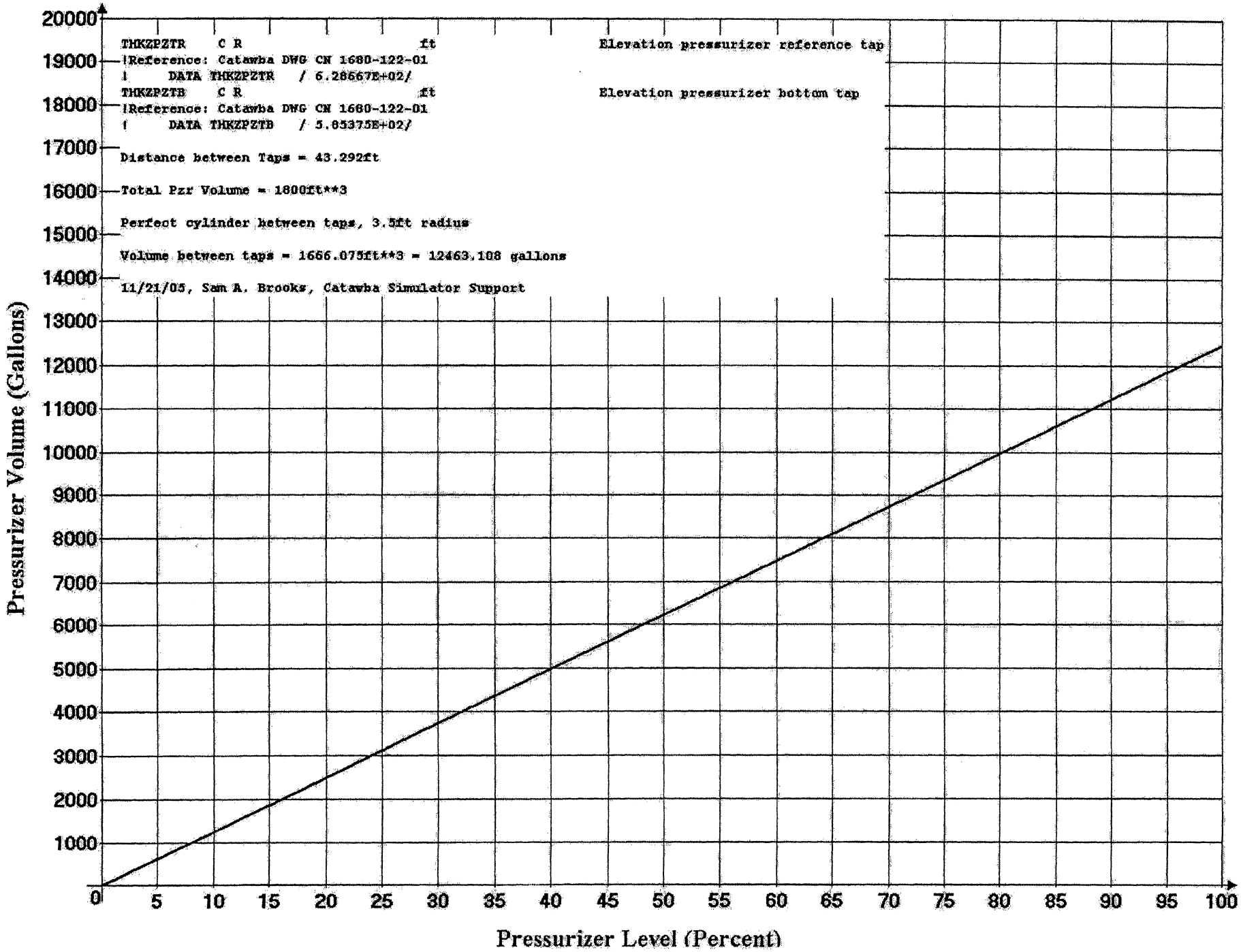
__ • Maintain S/I flow greater than or equal to the flow required by Enclosure 5 (Minimum S/I Flowrate Versus Time After Trip).

__ 3) **GO TO** Step 26.

S/I FLOW REQUIRED TO MATCH DECAY HEAT



Pressurizer Level Vs Volume



3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS — Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE*.

APPLICABILITY: MODES 1, 2, and 3.

----------NOTE----------
In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours*
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

*For Unit 1 only, the Completion Time that the 1B ECCS train can be inoperable as specified by Required Action A.1 may be extended beyond the "72 hours" up to a total of 240 hours as part of the 1B centrifugal charging pump repair. Upon completion of the repair and restoration, this footnote is no longer applicable and will expire at 0130 on January 10, 2008.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. Number / Position / Function NI162A / Open / SI Cold Leg Injection NI121A / Closed / SI Hot Leg Injection NI152B / Closed / SI Hot Leg Injection NI183B / Closed / RHR Hot Leg Injection NI173A / Open / RHR Cold Leg Injection NI178B / Open / RHR Cold Leg Injection NI100B / Open / SI Pump Suction from RWST NI147B / Open / SI Pump Mini-Flow	12 hours
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.3	Verify ECCS piping is full of water.	31 days
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. Centrifugal Charging Pump Injection Throttle Valve Number — Safety Injection Pump Throttle Valve Number NI14 — NI164 NI16 — NI166 NI18 — NI168 NI20 — NI170	18 months
SR 3.5.2.8	Verify, by visual inspection, that the ECCS containment sump strainer assembly is not restricted by debris and shows no evidence of structural distress or abnormal corrosion.	18 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE*:

a. Two qualified circuits between the offsite transmission network and the Onsite Essential Auxiliary Power System; and

b. Two diesel generators (DGs) capable of supplying the Onsite Essential Auxiliary Power Systems;

AND

The automatic load sequencers for Train A and Train B shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	

(continued)

*For each Unit, the Completion Time that one EDG can be inoperable as specified by Required Action B.4 may be extended beyond the "72 hours and 6 days from discovery of failure to meet the LCO" up to 336 hours as part of the NSWS system upgrades. System upgrades include maintenance activities associated with cleaning of NSWS piping; weld coating, and necessary repairs and/or replacement. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable and if not used, will expire at midnight on December 31, 2006.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Restore offsite circuit to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for the offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND	
	B.2 Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND	
	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore DG to OPERABLE status.	72 hours* AND 6 days* from discovery of failure to meet LCO
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours

(continued)

*For each Unit, the Completion Time that one EDG can be inoperable as specified by Required Action B.4 may be extended beyond the "72 hours and 6 days from discovery of failure to meet the LCO" up to 336 hours as part of the NSWS system upgrades. System upgrades include maintenance activities associated with cleaning of NSWS piping; weld coating, and necessary repairs and/or replacement. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable and if not used, will expire at midnight on December 31, 2006.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One offsite circuit inoperable. AND One DG inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems—Operating," when Condition D is entered with no AC power source to any train. ---------- D.1 Restore offsite circuit to OPERABLE status. OR D.2 Restore DG to OPERABLE status.	 12 hours 12 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours
F. One automatic load sequencer inoperable.	F.1 Restore automatic load sequencer to OPERABLE status.	12 hours
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours
H. Three or more AC sources inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each offsite circuit.	7 days
SR 3.8.1.2	----------NOTES---------- 1. Performance of SR 3.8.1.7 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.7 must be met. ---------- Verify each DG starts from standby conditions and achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	----------NOTES---------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. ---------- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 5600 kW and ≤ 5750 kW.	31 days
SR 3.8.1.4	Verify each day tank contains ≥ 470 gal of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage system to the day tank.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.7	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts from standby condition and achieves in ≤ 11 seconds voltage of ≥ 3740 V and frequency of ≥ 57 Hz and maintains steady-state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	184 days
SR 3.8.1.8	Verify automatic and manual transfer of AC power sources from the normal offsite circuit to each alternate offsite circuit.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	----------NOTE---------- If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. ---------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 63 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 3740 V and ≤ 4580 V; and c. Within 3 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.	18 months
SR 3.8.1.10	Verify each DG does not trip and generator speed is maintained ≤ 500 rpm during and following a load rejection of ≥ 5600 kW and ≤ 5750 kW.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes the emergency bus in ≤ 11 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage ≥ 3740 V and ≤ 4580 V, 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies auto-connected shutdown loads for ≥ 5 minutes.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	----------NOTE---------- All DG starts may be preceded by prelube period. ---------- Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and: a. In ≤ 11 seconds after auto-start and during tests, achieves voltage ≥ 3740 V and ≤ 4580 V; b. In ≤ 11 seconds after auto-start and during tests, achieves frequency ≥ 58.8 Hz and ≤ 61.2 Hz; c. Operates for ≥ 5 minutes; and d. The emergency bus remains energized from the offsite power system.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	Verify each DG's non-emergency automatic trips are bypassed on actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal.	18 months
SR 3.8.1.14	----------NOTE---------- Momentary transients outside the load and power factor ranges do not invalidate this test. ---------- Verify each DG operating at a power factor ≤ 0.9 operates for ≥ 24 hours loaded ≥ 5600 kW and ≤ 5750 kW.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	----------NOTES---------- 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 1 hour loaded ≥ 5600 kW and ≤ 5750 kW or until operating temperature is stabilized. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts and achieves, in ≤ 11 seconds, voltage ≥ 3740 V, and frequency ≥ 57 Hz and maintains steady state voltage ≥ 3740 V and ≤ 4580 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	18 months
SR 3.8.1.16	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to standby operation.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.17	----------NOTE---------- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to standby operation; and b. Automatically energizing the emergency load from offsite power.	18 months
SR 3.8.1.18	Verify interval between each sequenced load block is within the design interval for each automatic load sequencer.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.19	----------NOTES---------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ---------- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes the emergency bus in ≤ 11 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. achieves steady state voltage ≥ 3740 V and ≤ 4580 V, 4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies auto-connected emergency loads for ≥ 5 minutes.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.20	----------NOTE---------- All DG starts may be preceded by an engine prelube period. ---------- Verify when started simultaneously from standby condition, each DG achieves, in ≤ 11 seconds, voltage of ≥ 3740 V and frequency of ≥ 57 Hz and maintains steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	10 years

Enclosure 4.1

Fission Product Barrier Matrix

Use EALs to determine Fission Product Barrier status (Intact, Potential Loss, or Loss). Add points for all 3 barriers. Classify according to the table on page 2 of 5 of this enclosure.

Note 1: This table is only applicable in Modes 1-4.

Note 2: Also, an event (or multiple events) could occur which results in the conclusion that exceeding the Loss or Potential Loss thresholds is IMMINENT (i.e., within 1-3 hours). In this IMMINENT LOSS situation, use judgement and classify as if the thresholds are exceeded.

Note 3: When determining Fission Product Barrier status, the Fuel Clad Barrier should be considered to be lost or potentially lost if the conditions for the Fuel Clad Barrier loss or potential loss EALs were met previously (validated and sustained) during the event, even if the conditions do not currently exist.

Note 4: Critical Safety Function (CSF) indications are not meant to include transient alarm conditions which may appear during the start-up of engineered safeguards equipment. A CSF condition is satisfied when the alarmed state is **valid** and **sustained**. The STA should be consulted to affirm that a CSF has been validated prior to the CSF being used as a basis to classify an emergency.

Example: If ECA-0.0, Loss of All AC Power, is implemented with an appropriate CSF alarm condition **valid** and **sustained**, that CSF should be used as the basis to classify an emergency prior to any function restoration procedure being implemented within the confines of ECA-0.0.

EAL #	Unusual Event	EAL #	Alert	EAL #	Site Area Emergency	EAL #	General Emergency
4.1.U.1	Potential Loss of Containment	4.1.A.1	Loss **OR** Potential Loss of Nuclear Coolant System	4.1.S.1	Loss **OR** Potential Loss of Both Nuclear Coolant System **AND** Fuel Clad	4.1.G.1	Loss of All Three Barriers
4.1.U.2	Loss of Containment	4.1.A.2	Loss **OR** Potential Loss of Fuel Clad	4.1.S.2	Loss **AND** Potential Loss Combinations of Both Nuclear Coolant System **AND** Fuel Clad	4.1.G.2	Loss of Any Two Barriers **AND** Potential Loss of the Third
		4.1.A.3	Potential Loss of Containment **AND** Loss **OR** Potential Loss of Any Other Barrier	4.1.S.3	Loss of Containment **AND** Loss **OR** Potential Loss of Any Other Barrier		

Enclosure 4.1

Fission Product Barrier Matrix

NOTE: If a barrier is affected, it has a single point value based on a "potential loss" or a "loss." "Not Applicable" is included in the table as a place holder only, and has no point value assigned.

Barrier	Points (1-5)	Potential Loss (X)	Loss (X)	Total Points	Classification
Containment		1	3	1 – 3	Unusual Event
NCS		4	5	4 – 6	Alert
Fuel Clad		4	5	7 – 10	Site Area Emergency
Total Points				11 - 13	General Emergency

1. Compare plant conditions against the Fission Barrier Matrix on pages 3 through 5 of 5.
2. Determine the "potential loss" or "loss" status for each barrier (Containment, NCS and Fuel Clad) based on the EAL symptom description.
3. For each barrier, write the highest single point value applicable for the barrier in the "Points" column and mark the appropriate "loss" column.
4. Add the points in the "Points" column and record the sum as "Total Points".
5. Determine the classification level based on the number of "Total Points".
6. In the table on page 1 of 5, under one of the four "classification" columns, select the event number (e.g. 4.1.A.1 for Loss of Nuclear Coolant System) that best fits the loss of barrier descriptions.
7. Using the number (e.g. 4.1.A.1), select the preprinted notification form **OR** a blank notification form and complete the required information for Emergency Coordinator approval and transmittal.

Enclosure 4.1

Fission Product Barrier Matrix

4.1.C CONTAINMENT BARRIER

POTENTIAL LOSS - (1 Point)	LOSS – (3 Points)
1. Critical Safety Function Status	
• Containment-RED • Core cooling-RED Path is indicated for >15 minutes	• Not applicable
2. Containment Conditions	
• Containment Pressure > 15 PSIG • H2 concentration > 9% • Containment pressure greater than 3 psig with less than one full train of NS and a VX-CARF operating.	• Rapid unexplained decrease in containment pressure following initial increase • Containment pressure or sump level response not consistent with LOCA conditions.

CONTINUED

4.1.N NCS BARRIER

POTENTIAL LOSS - (4 Points)	LOSS – (5 Points)
1. Critical Safety Function Status	
• NCS Integrity-Red • Heat Sink-Red	• Not applicable
2. NCS Leak Rate	
• Unisolable leak exceeding the capacity of one charging pump in the normal charging mode with letdown isolated.	• GREATER THAN available makeup capacity as indicated by a loss of NCS subcooling.

CONTINUED

4.1.F FUEL CLAD BARRIER

POTENTIAL LOSS - (4 Points)	LOSS – (5 Points)
1. Critical Safety Function Status	
• Core Cooling-Orange • Heat Sink-Red	• Core Cooling-Red
2. Primary Coolant Activity Level	
• Not applicable	• Coolant Activity GREATER THAN 300 μCi/cc Dose Equivalent Iodine (DEI) I-131

CONTINUED

Enclosure 4.1

Fission Product Barrier Matrix

4.1.C CONTAINMENT BARRIER

POTENTIAL LOSS - (1 Point)	LOSS – (3 Points)
3. Containment Isolation Valves Status After Containment Isolation Actuation	
• Not applicable	• Containment isolation is incomplete and a release path from containment exists
4. SG Secondary Side Release With Primary-to-Secondary Leakage	
• Not applicable	• Release of secondary side to the environment with primary to secondary leakage GREATER THAN Tech Spec allowable

CONTINUED

4.1.N NCS BARRIER

POTENTIAL LOSS - (4 Points)	LOSS – (5 Points)
3. SG Tube Rupture	
• Primary-to-Secondary leak rate exceeds the capacity of one charging pump in the normal charging mode with letdown isolated.	• Indication that a SG is Ruptured and has a Non-Isolable secondary line fault • Indication that a SG is ruptured and a prolonged release of contaminated secondary coolant is occurring from the affected SG to the environment
4. Containment Radiation Monitoring	
• Not applicable	• Not applicable

CONTINUED

4.1.F FUEL CLAD BARRIER

POTENTIAL LOSS - (4 Points)	LOSS – (5 Points)
3. Containment Radiation Monitoring	
• Not applicable	• Containment radiation monitor 53 A or 53 B reading >117 R/hr
4. Emergency Coordinator/EOF Director Judgement	
• Any condition, including inability to monitor the barrier, that in the opinion of the Emergency Coordinator/EOF Director indicates **LOSS** or **POTENTIAL LOSS** of the fuel clad barrier.	

END

Enclosure 4.1

Fission Product Barrier Matrix

4.1.C CONTAINMENT BARRIER		4.1.N NCS BARRIER		4.1.F FUEL CLAD BARRIER	
POTENTIAL LOSS - (1 Point)	LOSS – (3 Points)	POTENTIAL LOSS - (4 Points)	LOSS – (5 Points)	POTENTIAL LOSS - (4 Points)	LOSS – (5 Points)
5. <u>Significant Radioactive Inventory In Containment</u> • Containment Rad. Monitor EMF53A or 53B Reading @ time since shutdown: > 470 R/hr @ 0 - 0.5 hr > 170 R/hr @ 0.5 - 2 hr > 125 R/hr @ 2 - 4 hr > 90 R/hr @ 4 - 8 hr > 53 R/hr @ > 8 hr	• Not applicable	**5. <u>Emergency Coordinator/EOF Director Judgement</u>** • Any condition, including inability to monitor the barrier, that in the opinion of the Emergency Coordinator /EOF Director indicates **LOSS** or **POTENTIAL LOSS** of the NCS barrier. **<u>END</u>**			
6. <u>Emergency Coordinator /EOF Director Judgement</u> • Any condition, including inability to monitor the barrier, that in the opinion of the Emergency Coordinator/EOF Director indicates **LOSS** or **POTENTIAL LOSS** of the containment barrier. **<u>END</u>**					

Enclosure 4.2

System Malfunctions

UNUSUAL EVENT

4.2.U.1 Inability to Reach Required Shutdown Within Technical Specification Limits.

OPERATING MODE: 1, 2, 3, 4

4.2.U.1-1 Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

4.2.U.2 Unplanned Loss of Most or All Safety System Annunciation or Indication in the Control Room for Greater Than 15 Minutes.

OPERATING MODE: 1 , 2, 3, 4

4.2.U.2-1 The following conditions exist:

Unplanned loss of most (>50%) annunciators associated with safety systems for greater than 15 minutes.

AND

In the opinion of the Operations Shift Manager/Emergency Coordinator/EOF Director, the loss of the annunciators or indicators requires additional personnel (beyond normal shift compliment) to safely operate the unit.

CONTINUED

ALERT

4.2.A.1 Unplanned Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Non-Alarming Indicators Unavailable.

OPERATING MODE: 1, 2, 3, 4

4.2.A.1-1 The following conditions exist:

Unplanned loss of most (>50%) annunciators associated with safety systems for greater than 15 minutes.

AND

In the opinion of the Operations Shift Manager/Emergency Coordinator/EOF Director, the loss of the annunciators or indicators requires additional personnel (beyond normal shift compliment) to safely operate the unit.

AND

EITHER of the following:

- A **significant plant transient** is in progress
- Loss of the OAC.

END

SITE AREA EMERGENCY

4.2.S.1 Inability to Monitor a Significant Transient in Progress.

OPERATING MODE: 1, 2, 3, 4

4.2.S.1-1 The following conditions exist:

Loss of most (>50%) Annunciators associated with safety systems.

AND

A **significant plant transient** is in progress.

AND

Loss of the OAC.

AND

Inability to provide manual monitoring of any of the following Critical Safety Functions:

- subcriticality
- core cooling
- heat sink
- containment.

END

GENERAL EMERGENCY

END

UNUSUAL EVENT | **ALERT** | **SITE AREA EMERGENCY** | **GENERAL EMERGENCY**

4.2.U.3 Fuel Clad Degradation.

OPERATING MODE: 1, 2, 3*

4.2.U.3-1 Dose Equivalent I-131 greater than the Technical Specifications allowable limit. (*Mode 3 with TAV >500° F)

4.2.U.4 Reactor Coolant System (NCS) Leakage.

OPERATING MODE: 1, 2, 3, 4

4.2.U.4-1 Unidentified leakage ≥ 10 gpm.

4.2.U.4-2 Pressure boundary leakage ≥ 10 gpm.

4.2.U.4-3 Identified leakage ≥ 25 gpm

4.2.U.5 Unplanned Loss of All Onsite or Offsite Communications.

OPERATING MODE: ALL

4.2.U.5-1 Loss of all onsite communications capability (internal phone system, PA system, onsite radio system) affecting the ability to perform routine operations.

4.2.U.5-2 Loss of all offsite communications capability (Selective Signaling, NRC ETS lines, offsite radio system, commercial phone system) affecting the ability to communicate with offsite authorities.

END

Enclosure 4.3

Abnormal Rad Levels/Radiological Effluent

UNUSUAL EVENT

4.3.U.1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the SLC Limits for 60 Minutes or Longer.

OPERATING MODE: ALL

4.3.U.1-1 A **valid** Trip 2 alarm on radiation monitor EMF-49L or EMF-57 for ≥ 60 minutes or will likely continue for ≥ 60 minutes which indicates that the release may have exceeded the initiating condition and indicates the need to assess the release with procedure HP/0/B/1009/014.

4.3.U.1-2 A **valid** indication on radiation monitor EMF- 36L of ≥ 3.00E+04 cpm for ≥ 60 minutes or will likely continue for ≥ 60 minutes, which indicates that the release may have exceeded the initiating condition and indicates the need to assess the release with procedure SH/0/B/2005/001.

(Continued)

ALERT

4.3.A.1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the SLC limits for 15 Minutes or Longer.

OPERATING MODE: ALL

4.3.A.1-1 A **valid** indication on radiation monitor EMF- 49L or EMF-57 of ≥ 1.2E+05 cpm for ≥ 15 minutes or will likely continue for ≥15 minutes, which indicates that the release may have exceeded the initiating condition and indicates the need to assess the release with procedure HP/0/B/1009/014.

(Continued)

SITE AREA EMERGENCY

4.3.S.1 Boundary Dose Resulting from an Actual or Imminent Release of Radioactivity Exceeds 100 mRem TEDE or 500 mRem CDE Adult Thyroid for the Actual or Projected Duration of the Release.

OPERATING MODE: ALL

4.3.S.1-1 A **valid** indication on radiation monitor EMF-36L of ≥ 2.7E+06 cpm **sustained** for ≥ 15 minutes.

4.3.S.1-2 Dose assessment team calculations indicate dose consequences greater than 100 mRem TEDE or 500 mRem CDE Adult Thyroid at the **site boundary**.

(Continued)

GENERAL EMERGENCY

4.3.G.1 Boundary Dose Resulting from an Actual or Imminent Release of Radioactivity that Exceeds 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid for the Actual or Projected Duration of the Release.

OPERATING MODE: ALL

4.3.G.1-1 A **valid** indication on radiation monitor EMF-36H of ≥ 8.3E+03 cpm **sustained** for ≥ 15 minutes.

4.3.G.1-2 Dose assessment team calculations indicate dose consequences greater than 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid at the **site boundary**.

(Continued)

Abnormal Rad Levels/Radiological Effluent

UNUSUAL EVENT

4.3.U.1-3 Gaseous effluent being released exceeds two times **SLC** 16.11-6 for ≥ 60 minutes as determined by RP procedure.

4.3.U.1-4 Liquid effluent being released exceeds two times **SLC** 16.11-1 for ≥ 60 minutes as determined by RP procedure.

Note: If the monitor reading is sustained for the time period indicated in the EAL AND the required assessments (procedure calculations) cannot be completed within this time period, declaration must be made based on the **valid** radiation monitor reading.

(Continued)

ALERT

4.3.A.1-2 A **valid** indication on radiation monitor EMF- 36L of ≥ 5.4E+05 cpm for ≥ 15 minutes or will likely continue for ≥15 minutes, which indicates that the release may have exceeded the initiating condition and indicates the need to assess the release with procedure SH/0/B/2005/001.

4.3.A.1-3 Gaseous effluent being released exceeds 200 times the level of **SLC** 16.11-6 for ≥ 15 minutes as determined by RP procedure.

4.3.A.1-4 Liquid effluent being released exceeds 200 times the level of **SLC** 16.11-1 for ≥ 15 minutes as determined by RP procedure.

Note: If the monitor reading is **sustained** for the time period indicated in the EAL AND the required assessments (procedure calculations) cannot be completed within this time period, declaration must be made based on the **valid** radiation monitor reading.

(Continued)

SITE AREA EMERGENCY

4.3.S.1-3 Analysis of field survey results or field survey samples indicates dose consequences greater than 100 mRem TEDE or 500 mRem CDE Adult Thyroid at the **site boundary**.

Note 1: These EMF readings are calculated based on average annual meteorology, site boundary dose rate, and design unit vent flow rate. Calculations by the dose assessment team use actual meteorology, release duration, and unit vent flow rate. Therefore, these EMF readings should not be used if dose assessment team calculations are available.

Note 2: If dose assessment team calculations cannot be completed in 15 minutes, then **valid** monitor reading should be used for emergency classification.

END

GENERAL EMERGENCY

4.3.G.1-3 Analysis of field survey results or field survey samples indicates dose consequences greater than 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid at the **site boundary**.

Note 1: These EMF readings are calculated based on average annual meteorology, site boundary dose rate, and design unit vent flow rate. Calculations by the dose assessment team use actual meteorology, release duration, and unit vent flow rate. Therefore, these EMF readings should not be used if dose assessment team calculations are available.

Note 2: If dose assessment team calculations cannot be completed in 15 minutes, then **valid** monitor reading should be used for emergency classification.

END

Enclosure 4.3

Abnormal Rad Levels/Radiological Effluent

UNUSUAL EVENT

4.3.U.2 Unexpected Increase in Plant Radiation or Airborne Concentration.

OPERATING MODE: ALL

4.3.U.2-1 Indication of **uncontrolled** water level decrease of greater than 6 inches in the reactor refueling cavity with all irradiated fuel assemblies remaining covered by water.

4.3.U.2-2 **Uncontrolled** water level decrease of greater than 6 inches in the spent fuel pool and fuel transfer canal with all irradiated fuel assemblies remaining covered by water.

4.3.U.2-3 **Unplanned valid** area EMF reading increases by a factor of 1000 over normal levels as shown in Enclosure 4.10.

END

ALERT

4.3.A.2 Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

OPERATING MODE: ALL

4.3.A.2-1 An **unplanned valid** trip II alarm on any of the following radiation monitors:

Spent Fuel Building Refueling Bridge
1EMF-15
2EMF-4

Spent Fuel Pool Ventilation
1EMF-42
2EMF-42

Reactor Building Refueling Bridge (applies to Mode 6 and No Mode Only)
1EMF-17
2EMF-2

Containment Noble Gas Monitor (Applies to Mode 6 and No Mode Only)
1EMF-39
2EMF-39

(Continued)

SITE AREA EMERGENCY

GENERAL EMERGENCY

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	4.3.A.2-2 Plant personnel report that water level drop in reactor refueling cavity, spent fuel pool, or fuel transfer canal has or will exceed makeup capacity such that any irradiated fuel will become uncovered.		
	4.3.A.2-3 NC system wide range level <95% after initiation of NC system make-up. **AND** Any irradiated fuel assembly not capable of being lowered into spent fuel pool or reactor vessel.		
	4.3.A.2-4 Spent Fuel Pool or Fuel Transfer Canal level decrease of >2 feet after initiation of makeup. **AND** Any irradiated fuel assembly not capable of being fully lowered into the spent fuel pool racks or transfer canal fuel transfer system basket. **(Continued)**		

UNUSUAL EVENT | **ALERT** | **SITE AREA EMERGENCY** | **GENERAL EMERGENCY**

4.3.A.3 Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown.

OPERATING MODE: ALL

4.3.A.3-1 **Valid** reading on EMF-12 greater than 15 mR/hr in the Control Room.

4.3.A.3-2 **Valid** indication of radiation levels greater than 15 mR/hr in the Central Alarm Station (CAS) or Secondary Alarm Station (SAS).

4.3.A.3-3 **Valid** radiation monitor reading exceeds the levels shown in Enclosure 4.10.

END

Enclosure 4.4

Loss of Shutdown Functions

UNUSUAL EVENT

END

ALERT

4.4.A.1 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was Successful.

OPERATING MODE: 1, 2, 3

4.4.A.1-1 The following conditions exist:

Valid reactor trip signal received or required and automatic reactor trip **was not** successful.

AND

Manual reactor trip from the control room is successful and reactor power is less than 5% and decreasing.

(Continued)

SITE AREA EMERGENCY

4.4.S.1 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Trip Was NOT Successful.

OPERATING MODE: 1

4.4.S.1-1 The following conditions exist:

Valid reactor trip signal received or required and automatic reactor trip **was not** successful.

AND

Manual reactor trip from the control room **was not** successful in reducing reactor power to less than 5% and decreasing.

(Continued)

GENERAL EMERGENCY

4.4.G.1 Failure of the Reactor Protection System to Complete an Automatic Trip and Manual Trip Was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

OPERATING MODE: 1

4.4.G.1-1 The following conditions exist:

Valid reactor trip signal received or required and automatic reactor trip **was not** successful.

AND

Manual reactor trip from the control room **was not** successful in reducing reactor power to less than 5% and decreasing.

AND

EITHER of the following conditions exist:

- Core Cooling CSF-RED
- Heat Sink CSF-RED.

END

Enclosure 4.4

Loss of Shutdown Functions

UNUSUAL EVENT

ALERT

4.4.A.2 Inability to Maintain Plant in Cold Shutdown.

OPERATING ODE: 5, 6

4.4.A.2-1 Total loss of ND and/or RN and/or KC.

AND

One of the following:

- Inability to maintain reactor coolant temperature below 200°F
- Uncontrolled reactor coolant temperature rise to >180°F.

END

SITE AREA EMERGENCY

4.4.S.2 Complete Loss of Function Needed to Achieve or Maintain Hot Shutdown.

OPERATING MODE: 1, 2, 3, 4

4.4.S.2-1 Subcriticality CSF-RED.

4.4.S.2-2 Heat Sink CSF-RED.

4.4.S.3 Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel.

OPERATING MODE: 5, 6

4.4.S.3-1 Failure of heat sink causes loss of cold shutdown conditions.

AND

Lower range Reactor Vessel Level Indication System (RVLIS) decreasing after initiation of NC system makeup.

4.4.S.3-2 Failure of heat sink causes loss of cold shutdown conditions.

AND

Reactor Coolant (NC) system mid or wide range level less than 11% and decreasing after initiation of NC system makeup.

(Continued)

GENERAL EMERGENCY

Enclosure 4.4

Loss of Shutdown Functions

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
		4.4.S.3-3 Failure of heat sink causes loss of cold shutdown conditions. **AND** Either train ultrasonic level indication less than 7.25% and decreasing after initiation of NC system makeup. **END**	

Enclosure 4.5

Loss of Power

UNUSUAL EVENT

4.5.U.1 Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes.

OPERATING MODE: 1, 2, 3, 4

4.5.U.1-1 The following conditions exist:

Loss of offsite power to essential buses ETA and ETB for greater than 15 minutes.

AND

Both emergency diesel generators are supplying power to their respective essential busses.

OPERATING MODE: 5, 6, No Mode

(Continued)

ALERT

4.5.A.1 Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses During Cold Shutdown Or Refueling Mode.

OPERATING MODE: 5, 6, No Mode

4.5.A.1-1 Loss of all offsite and onsite AC power as indicated by:

Loss of power on essential buses ETA and ETB.

AND

Failure to restore power to at least one essential bus within 15 minutes.

(Continued)

SITE AREA EMERGENCY

4.5.S.1 Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.

OPERATING MODE: 1, 2, 3, 4

4.5.S.1-1 Loss of all offsite and onsite AC power as indicated by:

Loss of power on essential buses ETA and ETB.

AND

Failure to restore power to at least one essential bus within 15 minutes.

4.5.S.2 Loss of All Vital DC Power.

OPERATING MODE: 1, 2, 3, 4

(Continued)

GENERAL EMERGENCY

4.5.G.1 Prolonged Loss of All (Offsite and Onsite) AC Power.

OPERATING MODE: 1, 2, 3, 4

4.5.G.1-1 Prolonged loss of all offsite and onsite AC power as indicated by:

Loss of power on essential buses ETA and ETB for greater than 15 minutes.

AND

Standby Shutdown Facility (SSF) fails to supply NC pump seal injection **OR** CA supply to Steam Generators.

AND

At least one of the following conditions exist:

- Restoration of at least one essential bus within 4 hours is **NOT** likely

(Continued)

Enclosure 4.5

Loss of Power

UNUSUAL EVENT

4.5.U.1-2 The following conditions exist:
Loss of offsite power to essential buses ETA and ETB for greater than 15 minutes.

AND

One emergency diesel generator is supplying power to its respective essential bus.

4.5.U.2 **Unplanned Loss of Required DC Power During Cold Shutdown or Refueling Mode for Greater than 15 Minutes.**

OPERATING MODE: 5, 6

4.5.U.2-1 The following conditions exist:

Unplanned loss of both unit related busses: EBA and EBD both <112 VDC, and EBB and EBC both <109 VDC.

AND

Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

END

ALERT

4.5.A.2 **AC power to essential busses reduced to a single power source for greater than 15 minutes such that an additional single failure could result in station blackout.**

OPERATING MODE: 1, 2, 3, 4

4.5.A.2-1 The following condition exists:

AC power capability has been degraded to one essential bus powered from a single power source for > 15 min. due to the loss of all but one of:

SATA	SATB
ATC	ATD
D/G A	D/G B

END

SITE AREA EMERGENCY

4.5.S.2-1 The following conditions exist:

Unplanned loss of both unit related busses: EBA and EBD both <112 VDC, and EBB and EBC both <109 VDC.

AND

Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

END

GENERAL EMERGENCY

- Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.

END

Enclosure 4.6

Fire/Explosion and Security Events

UNUSUAL EVENT

4.6.U.1 Fire Within Protected Area Boundary NOT Extinguished Within 15 Minutes of Detection OR Explosion Within the Protected Area Boundary.

OPERATING MODE: ALL

4.6.U.1-1 **Fire** in any of the following areas **NOT** extinguished within 15 minutes of control room notification or verification of a control room fire alarm.

- Reactor Building
- Auxiliary Building
- Diesel Generator Rooms
- Control Room
- RN Pumphouse
- SSF
- CAS
- SAS
- Doghouses
- FWST
- Turbine Building
- Service Building
- Interim Radwaste Building
- Equipment Staging Building.
- Monitor Tank Building
- ISFSI

(Continued)

ALERT

4.6.A.1 Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

OPERATING MODE: 1, 2, 3, 4, 5, 6

4.6.A.1-1 The following conditions exist: (Non-security events)
Fire or **explosion** in any of the following areas:

- Reactor Building
- Auxiliary Building
- Diesel Generator Rooms
- Control Room
- RN Pumphouse
- SSF
- CAS
- SAS
- FWST
- Doghouses (Applies in Mode 1, 2, 3, 4 only).

AND

One of the following:

- Affected safety system parameter indications show degraded performance

(Continued)

SITE AREA EMERGENCY

4.6.S.1 Security Event in a Plant Vital Area.

OPERATING MODE: ALL

4.6.S.1-1 Intrusion into any of the following plant areas by a **hostile force**:

- Reactor Building
- Auxiliary Building
- Diesel Generator Rooms
- Control Room
- RN Pumphouse
- SSF
- Doghouses
- CAS
- SAS.

4.6.S.1-2 Security confirmed **bomb** discovered/exploded in a **vital area**.

4.6.S.1-3 Security confirmed **sabotage** in a plant **vital area**.

4.6.S.1-4 Other security events as determined from Safeguards Contingency Plan and reported by the security shift supervision.

(Continued)

GENERAL EMERGENCY

4.6.G.1 Security Event Resulting in Loss of Physical Control of the Facility.

OPERATING MODE: ALL

4.6.G.1-1 A **HOSTILE FORCE** has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions.

END

Enclosure 4.6

Fire/Explosion and Security Events

UNUSUAL EVENT

4.6.U.1-2 Report by plant personnel of an unanticipated **explosion** within **protected area** boundary resulting in **visible damage** to permanent structure or equipment or a loaded cask in the ISFSI.

4.6.U.2 **Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.**

OPERATING MODE: All

4.6.U.2-1 Security events as determined from Safeguards Contingency Plan and reported by the security shift supervision.

4.6.U.2-2 A credible site-specific security threat notification.

4.6.U.2-3 A validated notification from NRC providing information of an aircraft threat.

4.6.U.2-4 **Hostage** situation/**extortion**.

4.6.U.2-5 A **violent civil disturbance** within the owner controlled area.

END

ALERT

- Plant personnel report **visible damage** to permanent structures or equipment within the specified area required to establish or maintain safe shutdown within the specifications.

Note: Only one train of a system needs to be affected or damaged in order to satisfy this condition.

4.6.A.2 **Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.**

OPERATING MODE: No Mode

4.6.A.2-1 The following conditions exist: (Non-security events)
Fire or **explosion** in any of the following areas:
- Spent Fuel Pool
- Auxiliary Building.
- RN Pumphouse

AND

One of the following:
- Spent Fuel Pool level and/or temperature show degraded performance

(Continued)

SITE AREA EMERGENCY

4.6.S.2 **Site Attack**

OPERATING MODE: ALL

4.6.S.2-1 A notification from the site security force that an armed attack, explosive attack, airliner impact, or other **HOSTILE ACTION** is occurring or has occurred within the protected area.

END

GENERAL EMERGENCY

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	• Plant personnel report **visible damage** to permanent structures or equipment supporting spent fuel pool cooling.		
	4.6.A.3 Other Security Events as Determined from (site-specific) Safeguards Contingency Plan.		
	OPERATING MODE: ALL		
	4.6.A.3-1 Other security events as determined from Safeguards Contingency Plan and reported by the security shift supervision.		
	4.6.A.4 Notification of an Airborne Attack Threat.		
	OPERATING MODE: ALL		
	4.6.A.4-1 A validated notification from NRC of airliner attack threat less than 30 minutes away.		

(Continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	4.6.A.5 Notification of HOSTILE ACTION within the OCA. **OPERATING MODE: ALL** **4.6.A.5-1** A notification from the site security force that an armed attack, explosive attack, airliner impact or other **HOSTILE ACTION** is occurring or has occurred within the OCA. **END**		

Enclosure 4.7

Natural Disasters, Hazards, And Other Conditions Affecting Plant Safety

UNUSUAL EVENT

4.7.U.1 Natural and Destructive Phenomena Affecting the Protected Area.

OPERATING MODE: ALL

4.7.U.1-1 Tremor felt and valid alarm on the "strong motion accelerograph".

4.7.U.1-2 Tremor felt and valid alarm on the "Peak shock annunciator".

4.7.U.1-3 Report by plant personnel of tornado striking within **protected area** boundary/ISFSI.

4.7.U.1-4 Vehicle crash into plant structures or systems within **protected area** boundary/ISFSI.

4.7.U.1-5 Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

4.7.U.1-6 Independent Spent Fuel Cask tipped over or dropped greater than 12 inches.

4.7.U.1-7 Uncontrolled flooding in the ISFSI area.

4.7.U.1-8 Tornado generated missiles(s) impacting the ISFSI.

(Continued)

ALERT

4.7.A.1 Natural and Destructive Phenomena Affecting the Plant Vital Area.

OPERATING MODE: ALL

4.7.A.1-1 **Valid** "OBE Exceeded" Alarm on 1AD-4,B/8

4.7.A.1-2 Tornado or high winds:

Tornado striking plant structures within the **vital area**:

- Reactor Building
- Auxiliary Building
- FWST
- Diesel Generator Rooms
- Control Room
- RN Pumphouse
- SSF
- Doghouses
- CAS
- SAS

OR

sustained winds ≥ 74 mph for > 15 minutes.

(Continued)

SITE AREA EMERGENCY

4.7.S.1 Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.

OPERATING MODE: ALL

4.7.S.1-1 The following conditions exist:
Control Room evacuation has been initiated per AP/1(2)/A/5500/017

AND

Control of the plant cannot be established from the ASP or the SSF within 15 minutes.

4.7.S.2 Other Conditions Existing Which in the Judgement of the Emergency Coordinator/EOF Director Warrant Declaration of Site Area Emergency.

OPERATING MODE: ALL

4.7.S.2-1 Other conditions exist which in the Judgement of the Emergency Coordinator/EOF Director indicate actual or likely major failures of plant functions needed for protection of the public.

END

GENERAL EMERGENCY

4.7.G.1 Other Conditions Existing Which in the Judgement of the Emergency Coordinator/EOF Director Warrant Declaration of General Emergency.

OPERATING MODE: ALL

4.7.G.1-1 Other conditions exist which in the Judgement of the Emergency Coordinator/EOF Director indicate:

(1) actual or **imminent** substantial core degradation with potential for loss of containment

OR

(2) potential for **uncontrolled** radionuclide releases. These releases can reasonably be expected to exceed Environmental Protection Agency Protective Action Guideline levels outside the site boundary.

END

Natural Disasters, Hazards, And Other Conditions Affecting Plant Safety

UNUSUAL EVENT

4.7.U.2 Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant.

OPERATING MODE: ALL

4.7.U.2-1 Report or detection of **toxic** or flammable **gases** that could enter within the **site boundary** in amounts that can affect safe operation of the plant.

4.7.U.2-2 Report by Local, County or State Officials for potential evacuation of site personnel based on offsite event.

4.7.U.3 Other Conditions Existing Which in the Judgement of the Emergency Coordinator/EOF Director Warrant Declaration of an Unusual Event.

OPERATING MODE: ALL

4.7.U.3-1 Other conditions exist which in the judgement of the Emergency Coordinator/EOF Director indicate a potential degradation of the level of safety of the plant.

END

ALERT

4.7.A.1-3 Turbine failure generated missiles, vehicle crashes or other catastrophic events causing visible structural damage on any of the following plant structures:

- Reactor Building
- Auxiliary Building
- FWST
- Diesel Generator Rooms
- Control Room
- RN Pumphouse
- SSF
- Doghouses
- CAS
- SAS

(Continued)

SITE AREA EMERGENCY

GENERAL EMERGENCY

Natural Disasters, Hazards, And Other Conditions Affecting Plant Safety

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	4.7.A.2 Release of Toxic or Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown.		
	OPERATING MODE: ALL		
	4.7.A.2-1 Report or detection of **toxic gases** within a Facility Structure in concentrations that will be life threatening to plant personnel.		
	4.7.A.2-2 Report or detection of flammable gases within a Facility Structure in concentrations that will affect the safe operation of the plant.		
	Structures for the above EALs: • Reactor Building • Auxiliary Building • Diesel Generator Rooms • Control Room • RN Pumphouse • SSF • CAS • SAS		

(Continued)

Natural Disasters, Hazards, And Other Conditions Affecting Plant Safety

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
	4.7.A.3 Control Room Evacuation Has Been Initiated. **OPERATING MODE: ALL** **4.7.A.3-1** Control Room evacuation has been initiated per AP/1(2)/A/5500/017. **4.7.A.4 Other Conditions Existing Which in the Judgement of the Emergency Coordinator/EOF Director Warrant Declaration of an Alert.** **OPERATING MODE: ALL** **4.7.A.4-1** Other conditions exist which in the Judgement of the Emergency Coordinator/EOF Director indicate that plant safety systems may be degraded and that increased monitoring of plant functions is warranted. **END**		

SRO KEY CATAWBA 2008-301

KEY ID
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T F
1 A B C ● E
2 A B C ● E
3 A B C ● E
4 ● B C D E
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68 A B C ● E
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T F
76 A B C ● E
77 A B ● ● E Changed to "D"
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79 A ● C D E
80 A B C ● E
81 A ● C D E
82 A ● C D E
83 A B C ● E
84 ● B C D E
85 ● B C D E
86 A ● C D E
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88 A B C ● E
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90 A B C ● E
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93 A B C ● E
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95 A B C ● E
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98 A B ● D E
99 A B ● D E
100 ● B C D E

ANSWER KEY INFO.
OF KEYS
ITEM COUNT

PERFORMANCE ASSESSMENT
% OF TOTAL SCORE
00 = 100%
EQUAL PT VALUE ○
POINTS EARNED

Bar Code

FEED IN THIS DIRECTION

NUMBER CORRECT
PERCENT CORRECT
ROSTER NUMBER
SCORE
RESCORE

COMBINED POINTS EARNED
COMBINED PERCENT CORRECT
LETTER GRADE
SCORE
RESCORE

100 ITEM

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PERIOD
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A ● C D E
Fill oval completely
A B C D E
Erase cleanly

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32	A	B	C	D	E	
33	A	B	C	D	E	
34	A	B	C	D	E	
35	A	B	C	D	E	
36	A	B	C	D	E	
37	A	B	C	D	E	
38	A	B	C	D	E	
39	A	B	C	D	E	
40	A	B	C	D	E	
41	A	B	C	D	E	
42	A	B	C	D	E	
43	A	B	C	D	E	
44	A	B	C	D	E	
45	A	B	C	D	E	
46	A	B	C	D	E	
47	A	B	C	D	E	
48	A	B	C	D	E	
49	A	B	C	D	E	
50	A	B	C	D	E	
51	A	B	C	D	E	
52	A	B	C	D	E	
53	A	B	C	D	E	
54	A	B	C	D	E	
~~55~~	~~A~~	~~B~~	~~C~~	~~D~~	~~E~~	Dg
56	A	B	C	D	E	
57	A	B	C	D	E	
58	A	B	C	D	E	
59	A	B	C	D	E	
60	A	B	C	D	E	
61	A	B	C	D	E	
62	A	B	C	D	E	
63	A	B	C	D	E	
64	A	B	C	D	E	
65	A	B	C	D	E	
66	A	B	C	D	E	
67	A	B	C	D	E	
68	A	B	C	D	E	
69	A	B	C	D	E	
70	A	B	C	D	E	
71	A	B	C	D	E	
72	A	B	C	D	E	
73	A	B	C	D	E	
74	A	B	C	D	E	
75	A	B	C	D	E	
76	A	B	C	D	E	
77	A	B	~~C~~	D	E	D is correct
78	A	B	C	D	E	
79	A	B	C	D	E	
80	A	B	C	D	E	
81	A	B	C	D	E	
82	A	B	C	D	E	
83	A	B	C	D	E	
84	A	B	C	D	E	
85	A	B	C	D	E	
86	A	B	C	D	E	
~~87~~	~~A~~	~~B~~	~~C~~	~~D~~	~~E~~	Dg
88	A	B	C	D	E	
~~89~~	~~A~~	~~B~~	~~C~~	~~D~~	~~E~~	Dg
90	A	B	C	D	E	
91	A	B	C	D	E	
92	A	B	C	D	E	
93	A	B	C	D	E	
94	A	B	C	D	E	
95	A	B	C	D	E	
96	A	B	C	D	E	
97	A	B	C	D	E	
98	A	B	C	D	E	
99	A	B	C	D	E	
100	A	B	C	D	E	

ANSWER KEY INFO.			
# OF KEYS			
ITEM COUNT			
0	0	0	2
1	1	1	3
	2	2	4
	3	3	
	4	4	
	5	5	
	6	6	
	7	7	
	8	8	
	9	9	

PERFORMANCE ASSESSMENT					
	% OF TOTAL SCORE			POINTS EARNED	
	00 = 100%				
EQUAL PT VALUE	0	0	0	0	0
	1	1	1	1	1
	2	2	2	2	2
	3	3		3	3
	4	4		4	4
	5	5		5	5
	6	6		6	6
	7	7		7	7
	8	8		8	8
	9	9		9	9

Bar Code

FEED IN THIS DIRECTION

NUMBER CORRECT	
PERCENT CORRECT	
ROSTER NUMBER	
SCORE	
RESCORE	

PEARSON NCS

COMBINED POINTS EARNED	
COMBINED PERCENT CORRECT	
LETTER GRADE	
SCORE	
RESCORE	

100 ITEM

NAME Catawba 2008 Initial SRO
SUBJECT Exam Key
PERIOD DATE

MARKING INSTRUCTIONS

Use a No. 2 Pencil

(A) ● (C) (D) (E)
Fill oval completely

(A) (B) (C) (D) (E)
Erase cleanly

STUDENT ID NUMBER

0	0	0	0	0	0	0	0	0	0
1	1	1	1	1	1	1	1	1	1
2	2	2	2	2	2	2	2	2	2
3	3	3	3	3	3	3	3	3	3
4	4	4	4	4	4	4	4	4	4
5	5	5	5	5	5	5	5	5	5
6	6	6	6	6	6	6	6	6	6
7	7	7	7	7	7	7	7	7	7
8	8	8	8	8	8	8	8	8	8
9	9	9	9	9	9	9	9	9	9

Examination KEY for: 2008 SRO NRC

QUESTION 1-75 RO QUESTIONS
76-100 - SRO QUESTIONS

Question Number	*Answer*	
1	D	
2	D	
3	D	
4	A	
~~5~~	~~D~~	DELETED
6	A	
7	B	
8	D	
9	C	
10	B	
11	C	
12	A	
13	B	
14	D	
15	D	
16	D	
17	C	
18	C	
19	D	
20	B	
21	C	
22	A	
23	D	
24	D	
25	C	

Examination KEY for: 2008 SRO NRC

Question Number	*Answer*
26	B
27	B
28	D
29	B
30	C
31	D
32	C
33	D
34	A
35	A
36	B
37	D
38	C
39	A
40	D
41	A
42	C
43	A
44	D
45	C
46	D
47	B
48	C
49	D
50	B

Examination KEY for: 2008 SRO NRC

Question Number	*Answer*
51	A
52	B
53	A
54	B
~~55~~	~~C~~ DELETED
56	D
57	D
58	C
59	A
60	C
61	C
62	B
63	A
64	C
65	A
66	D
67	B
68	D
69	A
70	B
71	B
72	B
73	C
74	B
75	A

Examination KEY for: 2008 SRO NRC

Question Number	*Answer*
76	D
77	~~C~~ D changed to D
78	B
79	B
80	D
81	B
82	B
83	D
84	A
85	A
86	B
~~87~~	~~C~~ DELETED
88	D
~~89~~	~~B~~ DELETED
90	D
91	A
92	C
93	D
94	A
95	D
96	B
97	B
98	C
99	C
100	A